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Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement

Volume 1
Appendix A

Hanford Site
Spent Nuclear Fuel Management Program

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1. INTRODUCTION

The U.S. Department of Energy (DOE) is currently deciding the direction of its environmental restoration and waste management programs at the Idaho National Engineering Laboratory (INEL) for the next 10 years. Pertinent to this decision is establishing policies for the environmentally sensitive and safe transport, storage, and management of spent nuclear fuels (SNF). To develop these policies, it is necessary to revisit or examine the available options.

As a part of the DOE complex, the Hanford Site not only has a large portion of the nationwide DOE-owned inventory of SNF, but also is a participant in the DOE decision for management and ultimate disposition of SNF. Efforts in this process at Hanford include assessment of several options for stabilizing, transporting, and storing all or portions of DOE-owned SNF at the Hanford Site. Such storage and management of SNF will be in a safe and suitable manner until a final decision is made for ultimate disposition of SNF. The Hanford Site will be affected by the alternative chosen.

Five alternatives involving the Hanford Site are being considered for management of the SNF inventory: 1) the No Action Alternative, 2) the Decentralization Alternative, 3) the 1992/1993 Planning Basis Alternative, 4) the Regionalization Alternative, and 5) the Centralization Alternative. All alternatives will be carefully designed to avoid environmental degradation and to provide protection to human health and safety at the Hanford Site and surrounding region. For Hanford, these alternatives are briefly summarized below:

- No Action Alternative -- The No Action Alternative would preclude any additional transportation of SNF to or from Hanford but could include activities to maintain safe and secure materials and facilities. Hanford SNF would continue to be managed in the current mode and upgrade of existing facilities would occur only as required to ensure safety and security.

- Decentralization Alternative -- The Decentralization Alternative would require that DOE-owned fuel be managed at the location where it is removed from the reactor. Hanford SNF would be safely stored, with some limited onsite relocation of SNF. To accommodate this mission, existing facilities would be upgraded and new storage systems would be constructed.

- 1992/1993 Planning Basis -- SNF would continue to be managed in the current mode, which includes upgrades, fuel stabilization, transport of some SNF to either INEL or Savannah River Site for storage, and construction of an SNF storage facility at Hanford.

- Regionalization Alternative -- The Regionalization Alternative contains options that range from storing all SNF west of the Mississippi River including Naval SNF, to shipping all Hanford SNF offsite to either INEL or the Nevada Test Site. Existing facilities would be upgraded and new storage systems constructed, as in the Decentralization Alternative for SNF storage at Hanford, or packaging facilities would be constructed as in the Centralization (Minimum) Alternative for off-site shipment.

- Centralization Alternative -- The Centralization Alternative has two major options. Either all Hanford SNF would be shipped offsite to another location where all SNF would be centralized (minimum option), or the Hanford Site would become the centralized location (maximum option) for all DOE SNF to be stored until ultimate disposition.

The Spent Fuel Working Group Report (DOE 1993a) identified deficiencies related to existing SNF management at the various DOE sites. Most of these deficiencies result from degradation of the fuel and the facilities that store fuel because of the age of these facilities and the fuel storage conditions. Corrective actions to the identified deficiencies for each site, including the Hanford Site, are listed in DOE (1994a). Hanford Site corrective actions important to this EIS include the following:

1. alternative containerization of fuel stored in the 105-KE Basin to isolate a potential pathway of fuel constituents to the environment

2. preparation of a K Basins EIS and issuance of the record of decision to provide for management of SNF in the K Basins at the Hanford Site (SNF storage siting and configuration, path forward for ultimate disposition, etc.)

3. removal of all fuel and sludge from the K Basins by December 2002 based on the K Basins EIS record of decision

4. technical evaluation and characterization of N Reactor fuel to support development of the K Basins EIS

5. removal of fuel from the Fast Flux Test Facility; the Plutonium and Uranium Recovery through EXtraction (PUREX) Plant; the 308 Building; the 324, 325, and 327 buildings; T Plant; and the 200-West Area Low-Level Burial Grounds to support prolonged safe, economic, environmentally sound management of those fuels.

On-going corrective actions with prior National Environmental Policy Act (NEPA) coverage, such as containerization of fuel in the 105-KE Basin, are included in the No Action Alternative. Other corrective actions are included within the scope of each of the remaining alternatives. The impacts of continued fuel and facility degradation in the No Action

Alternative are not fully quantified, although it is generally recognized that prolonged storage in the existing facilities for an additional 40-year period might represent unacceptable risks, as reflected in DOE (1993a).

The Hanford Site portion of this EIS was prepared according to the National Environmental Policy Act (NEPA) of 1969, as amended; the Council on Environmental Quality (CEQ) regulations (40 CFR Part 1500-1508) for the implementation of the NEPA; and DOE regulations (10 CFR 1021) that supplement the CEQ regulations. This document discusses five alternatives for the management and storage of SNF, the affected environment, and potential impacts of the alternatives.

2. BACKGROUND

2.1 Hanford Site Overview

2.1.1 Site Description

The U.S. Department of Energy's Hanford Site lies within the semiarid Pasco Basin of the Columbia Plateau in southeastern Washington State (Figure 2.1). The Hanford Site occupies an area of about 1450 square kilometers (560 square miles) north of the confluence of the Yakima River with the Columbia River. The Hanford Site is about 50 kilometers (30 miles) north to south and 40 kilometers (24 miles) east to west. This land, with restricted public access, provides a buffer for the smaller areas previously used for production of nuclear materials, and currently used for research, waste management and disposal, and environmental restoration; only about 6 percent of the land area has been disturbed and is actively used. The Columbia River flows through the northern part of the Hanford Site, and turning south, it forms part of the site's eastern boundary. The Yakima River runs near the southern boundary and joins the Columbia River south of the city of Richland, which bounds the Hanford Site on the southeast. Rattlesnake Mountain, the Yakima Ridge, and the Umptanum Ridge form the southwestern and western boundary. The Saddle Mountains form the northern boundary of the Hanford Site. Two small east-west ridges, Gable Butte and Gable Mountain, rise above the plateau of the central part of the Hanford Site. Underneath the Hanford Site are ancient basaltic flows with basaltic outcroppings on the surface and intermixed beds of sand and gravel from ancient periods of flooding and glacial epochs. Adjoining lands to the west, north, and east are principally range and agricultural land. The cities of Richland, Kennewick, and Pasco (Tri-Cities) constitute the nearest population center and are located southeast of the Hanford Site.

The Hanford Site is listed on the National Priorities List under the Comprehensive Environmental Response, Compensation, and Liability Act. The site encompasses more than 1500 waste management units and four groundwater contamination plumes that have been grouped into 78 operable units. Each unit has complementary characteristics of such parameters as geography, waste characteristics, type of facility, and relationship of contaminant plumes. This grouping into operable units allows for economies of scale to reduce the cost and the number of characterization investigations and remedial actions that will be required for the

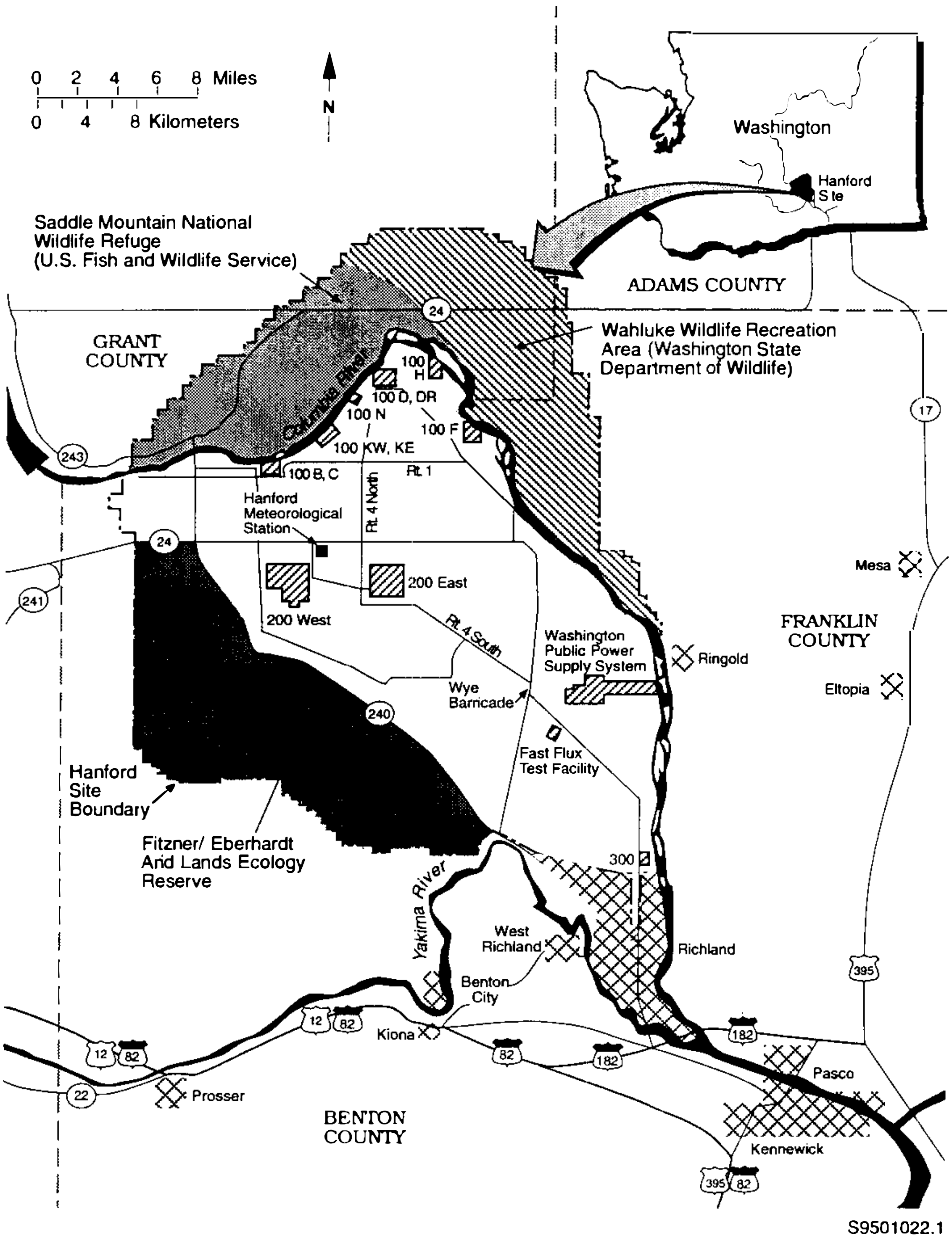

Figure 2-1. Hanford Site and vicinity.

Hanford Site to complete cleanup efforts. More information on the locations of the units is included in Section 4.1. Current maps showing the locations of the operable units can be obtained from Westinghouse Hanford Company.

2.1.2 History

The Hanford Site was acquired by the federal government in 1943. For more than 20 years, Hanford Site facilities were dedicated primarily to the production of plutonium for national defense and to the management of the resulting wastes. In later years, programs at the Hanford Site were diversified to include research and development for advanced reactors, renewable energy technologies, waste disposal technologies, and cleanup of contamination from past practices.

2.1.3 Mission

The new mission for Hanford emphasizes these components:

- Waste management of stored defense wastes and the handling, storage, and disposal of radioactive, hazardous, mixed, or sanitary wastes from current operations.
- Environmental restoration of approximately 1,500 inactive radioactive, hazardous, and mixed-waste sites and about 100 surplus facilities.
- Research and development in energy, health, safety, environmental sciences, molecular sciences, environmental restoration, and waste management.
- Technology development of new environmental restoration and waste management technologies, including site characterization and assessment methods; waste minimization, treatment, and remediation technology; and education outreach programs.

The DOE has set a goal of cleaning up Hanford's waste sites and bringing its facilities into compliance with local, state, and federal environmental laws by 2018.

2.1.4 Management

The Hanford Site is owned by the federal government and managed by the U.S. Department of Energy, Richland Operation's Office (DOE-RL). Westinghouse Hanford Company is the site operations and engineering contractor. Pacific Northwest Laboratory, which is operated for the DOE by Battelle Memorial Institute, manages the research and technology laboratories. In 1994, Bechtel Hanford Company and a team of contractors became DOE's environmental restoration contractor at the Hanford Site.

2.2 Regulatory Framework

The policy of DOE-RL is to carry out its operations in compliance with all applicable federal laws and regulations, state laws and regulations, presidential executive orders, and DOE orders. Environmental regulatory authority over the Hanford Site is vested both in federal agencies, primarily the U.S. Environmental Protection Agency (EPA), and in Washington State agencies, primarily the Department of Ecology. Significant environmental laws and regulations relevant to the management of SNF at Hanford are discussed in this section. First, major relevant federal and Washington State statutes are listed. Next, the specific topical concerns associated with spent nuclear fuel are discussed with appropriate citations to federal and state statutes and regulations. U.S. Department of Energy Orders will not be cited in this discussion because DOE Orders are not regulations. However, DOE Orders do delineate specific DOE procedures and provide detailed internal guidance for implementation of federal environmental, safety, and health regulations. DOE Orders establish specific standards, rules, and requirements that supplement the federal regulations for the design and construction of new facilities, and the operation of existing facilities to ensure safe and environmentally sound operations. Finally, it should be noted that environmental restoration and waste management activities at Hanford are governed by the Hanford Federal Facility Agreement and Consent Order (Tri-Party Agreement), which includes detailed provisions for state and federal jurisdiction, as well as specific goals for site management and cleanup. The Fourth Amendment to the Tri-Party Agreement (January 1994) contains specific milestones (M-34) related to the management of SNF at the Hanford Site.

2.2.1 Significant Federal and State Laws

Significant federal and state environmental and nuclear materials management laws applicable to the Hanford Site include the following (grouped by federal and state and listed alphabetically):

Federal Laws

- American Antiquities Act (16 U.S.C. 431-433)

- American Indian Religious Freedom Act (42 U.S.C. 1996)
- Archaeological and Historic Preservation Act (16 U.S.C. 469-469c)
- Archaeological Resources Protection Act (16 U.S.C. 470aa-470ll)
- Atomic Energy Act (AEA) (42 U.S.C. 2011 et seq.)
- Bald and Golden Eagle Protection Act (16 U.S.C 668-668d)
- Clean Air Act (CAA) as amended by the Clean Air Act Amendments of 1990 (42 U.S.C. 7401 et seq.)
- Clean Water Act (CWA) (33 U.S.C. 1251 et seq.)
- Comprehensive Conservation Study of the Hanford Reach of the Columbia River (PL 100-605)
- Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) as amended by the Superfund Amendments and Reauthorization Act (SARA) (42 U.S.C. 9601 et seq.)
- Emergency Planning and Community Right-to-Know Act of 1986 (42 U.S.C. 11001 et seq.)
- Endangered Species Act (16 U.S.C. 1531-1534)
- Energy Reorganization Act of 1974 (ERA) (42 USC 5801 et seq.)
- Federal Facilities Compliance Act (PL 102-386)
- Fish and Wildlife Coordination Act (16 U.S.C. 661-666c)
- Hazardous Materials Transportation Act (HMTA) (49 USC 1801 et seq.)
- Migratory Bird Treaty Act (16 U.S.C 703-711)
- National Environmental Policy Act (NEPA) (42 U.S.C. 4321 et seq.)
- National Historic Preservation Act (16 U.S.C. 470-470w-6)
- Native American Graves Protection and Repatriation Act (NAGPRA) (25 U.S.C. 3001 et seq.)
- Nuclear Waste Policy Act (NWPA) (42 U.S.C. 10101 et seq.)
- Pollution Prevention Act of 1990 (42 U.S.C. 13101 et seq.)
- Resource Conservation and Recovery Act (RCRA) as amended by the Hazardous and Solid Waste Amendments (42 U.S.C. 6901 et seq.)
- Safe Drinking Water Act (SDWA) (42 U.S.C. 300f et seq.)

- Toxic Substances Control Act (15 U.S.C. 2601 et seq.)
- Wild and Scenic Rivers Act (16 U.S.C. 1274 et seq.)

State Laws

- Washington Archaeological and Historic Preservation Code (RCW Chapter 27.34 et seq.)
- Washington Clean Air Act of 1967 (RCW Chapter 70.94 et seq.)
- Washington Hazardous Waste Management Act of 1976 (RCW Chapter 70.105 et seq.)
- Washington Model Toxics Control Act (RCW Chapter 70.105D).
- Washington Water Pollution Control Act (RCW 90.48 et seq.).

2.2.2 Environmental Standards for Spent Nuclear Fuel Storage Facilities

Design and performance standards for the construction and operation of SNF storage facilities arise from the Atomic Energy Act, Nuclear Waste Policy Act, Clean Water Act, and Clean Air Act, parallel state implementation statutes, and other major environmental/nuclear activities statutes. A general listing of regulations promulgated under these authorities will not be included in this discussion of the regulatory framework; relevant regulations will be cited as appropriate in the topical discussions that follow.

2.2.2.1 General Environmental Requirements for Construction and Operation. Design and construction of new facilities, modification of existing facilities, and operation of all facilities would be conducted in accordance with applicable state and federal environmental regulations. Special consideration with respect to operations of SNF management facilities at Hanford are discussed in the following sections.

Columbia River water would be used to serve a wet SNF storage facility. The DOE has asserted that it has federally reserved water withdrawal rights with respect to its Hanford operations. Nevertheless, DOE submitted an application to the Washington State Department of Ecology on July 7, 1987, as a matter of comity for water withdrawal rights from the Columbia River for site characterization activities related to the now defunct Basalt Waste Isolation Project. It may be appropriate to maintain this protocol with Washington State in regard to future withdrawals from the river.

Operation of SNF facilities may involve the generation of waste materials or unintentional releases of waste materials to the environment. The Pollution Prevention Act requires prevention or reduction of waste at the source whenever feasible. Reporting and cleanup of spills from an SNF facility are governed by CERCLA regulations (40 CFR 300, "National Oil and Hazardous Substances Pollution Contingency Plan"), which apply to the release of hazardous substances into the environment, including radioactive substances.

Shipment of SNF is governed by Department of Transportation hazardous materials regulations in 49 CFR 171-179 (under the authority of the Hazardous Materials Transportation Act), which apply to the handling, packaging, labeling, and shipment of hazardous materials offsite, including radioactive materials and wastes. Safety standards for packaging and transporting radioactive materials are governed by U.S. Nuclear Regulatory Commission (NRC) standards established in 10 CFR Part 71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions."

2.2.2.2 Resource Conservation and Recovery Act. The status of SNF with respect to RCRA is discussed in Volume 1. Most of the authority to administer the RCRA program, including treatment, storage and disposal standards, and permit requirements, has been delegated by EPA to the State of Washington, except for corrective action (cleanup). Washington State RCRA (WSHWMA) Dangerous Waste Regulations are found in WAC 173-303 (Washington Administrative Code). Generally, RCRA does not apply to source material, special nuclear material, by-product material, SNF, or radioactive-only wastes. Should SNF be processed into or commingled with a hazardous waste as defined by Subtitle C of RCRA, then the generation, treatment, storage, and disposal of the hazardous waste portion of such mixed waste would be subject to EPA regulations in 40 CFR 260-268 and 270-272.

2.2.2.3 Effluents. Regulations in 40 CFR 122 (and also in 40 CFR 125 and 129) apply to the discharge of pollutants from any point source into waters of the United States. A National Pollutant Discharge Elimination System (NPDES) permit is required for such discharges, which would include any effluent discharge from an SNF storage facility into the Columbia River. The EPA has not yet delegated to the State of Washington the authority to issue NPDES permits at the Hanford Site. At 40 CFR 121 the regulations provide for state certification that any activity requiring a federal CWA water permit, i.e., an NPDES permit or a discharge of dredged or fill material permit, will not violate state water quality standards.

The EPA drinking water standards in 40 CFR 141, "National Primary Drinking Water Regulations," apply to Columbia River water at community water supply intakes downstream of the Hanford Site. Washington Administrative Code 173-200 sets water quality standards for groundwater, and WAC 173-201 establishes surface water quality standards for the State of Washington.

Department of Ecology regulations in WAC 173-216 establish a state permit program, commonly referred to as the 216 program, for the discharge of waste materials from industrial, commercial, and municipal operations into ground and surface waters of the state. Discharges covered by NPDES or WAC 173-218 (Underground Injection Control Program) permits are excluded from the 216 program. The DOE has agreed to meet the requirements of the 216 program at the Hanford Site for discharges of liquids to the ground.

2.2.2.4 Air Quality. Hazardous emission standards in 40 CFR 61, "National Emission Standards for Hazardous Air Pollutants," provide for the control of the emission of hazardous pollutants to the atmosphere, and standards in 40 CFR 61, Subpart H, "National Emission Standards for Emissions of Radionuclides Other Than Radon from Department of Energy Facilities," apply specifically to the emission of radionuclides from DOE facilities. Approval to construct a new facility or to modify an existing one may be required by these regulations. The EPA has not yet delegated this approval authority to the State of Washington for the Hanford Site.

The Clean Air Act Amendments of 1990 require the addition of 189 substances to the list of hazardous air pollutants to be regulated on a schedule that extends to 1999. The hazardous air pollutant list includes radionuclides. The amendments require the identification of source categories and the definition of required control technology (maximum available control technology) for each of these pollutants. Hanford may fall within the definition of a major source because total emissions from Hanford may exceed the triggering limit of 25 tons per year for any combination of listed hazardous air pollutants (emission standards using curies as the unit of measure for radionuclides will be promulgated in the future). This means that emission sources at Hanford may become subject to permitting and reporting requirements and to installation requirements (including retrofit) for control technology. A new SNF storage facility may be subject to the maximum available control technology requirements for new sources.

Washington State Department of Health regulations in WAC 246-247, "Monitoring and Enforcement of Air Quality and Emission Standards for Radionuclides," contain standards and permit requirements for the emission of radionuclides to the atmosphere from DOE facilities based on Department of Ecology standards in WAC 173-480, "Ambient Air Quality Standards and Emission Limits for Radionuclides."

The local air authority, Benton County Clean Air Authority, enforces General Regulation 80-7, which pertains to detrimental effects, fugitive dust, incineration products, odor, opacity, asbestos, and sulfur oxide emissions. Benton County Clean Air Authority has been delegated authority to enforce EPA asbestos regulations.

2.2.3 Protection of Public Health

Numerical standards for protection of the public from releases to the environment have been set by the EPA and appear in the Code of Federal Regulations. The most significant of the regulations are discussed in the following paragraphs.

Clean Air Act standards found in 40 CFR 61.92 apply to releases of radionuclides to the atmosphere from DOE facilities and state as follows:

> Emissions of radionuclides [other than radon-220 and radon-222] to the ambient air from Department of Energy facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 millirem/year.

Safe Drinking Water standards found in 40 CFR 141.16 apply indirectly to releases of radionuclides from DOE facilities to the extent that the releases impact community water systems:

> The average annual concentration of beta particle and photon radioactivity from man-made radionuclides in drinking water shall not produce an annual dose equivalent to the body or any internal organ greater than 4 millirem/year.

Also, maximum contaminant levels in community water systems of 5 picocuries per liter of combined radium-226 and radium-228, and maximum contaminant levels of 15 picocuries per liter of gross alpha particle activity, including radium-226 but excluding radon and uranium, are specified in 40 CFR 141. The tritium concentration that corresponds to a dose of 4 millirem per year is 20,000 picocuries per liter.

2.2.4 Species Protection

Regulations of the Endangered Species Act, the Bald and Golden Eagle Protection Act, and the Migratory Bird Treaty Act in 50 CFR 10-24, 222, 225-227, 402, and 450-453 apply to the Hanford Site. The Endangered Species Act requires a biological assessment to identify any threatened or endangered species likely to be affected by the proposed action.

2.2.5 Floodplains and Wetlands

Executive Order 11988, "Floodplain Management," Executive Order 11990, "Protection of Wetlands," and 10 CFR 1022, require an assessment of the effects of DOE actions on floodplains and wetlands. These requirements are directed at the protection of water quality and habitat.

2.2.6 Cultural and Historic Preservation

Requirements of the National Historic Preservation Act in 36 CFR 800, the American Antiquities Act in 25 CFR 261 and 43 CFR 3, and the Archaeological Resources Protection Act and the American Indian Religious Freedom Act in 43 CFR 7 apply to the protection of historic and cultural properties, including both existing properties and those discovered during excavation and construction. The American Indian Religious Freedom Act and the Native American Graves Protection and Repatriation Act also provide for certain rights of access by Native Americans to traditional areas of worship and religious significance.

2.3 Spent Nuclear Fuel Management Program

This section presents a summary of current plans, as of December 1994, for the management of existing SNF on the Hanford site. The following SNF and associated facilities are at Hanford (Bergsman 1994):

- N Reactor SNF- Zircaloy-clad metallic uranium fuel stored in water in the 105-KW and 105-KE basins and exposed to air in the Plutonium and Uranium Recovery through Extraction (PUREX) Plant dissolver cells A, B, and C.

- Single-pass reactor SNF - aluminum-clad metallic uranium fuel stored in water in the 105-KE and 105-KW basins and stored in water in the PUREX basin.

- Shippingport Core II SNF - Zircaloy-clad uranium dioxide fuel stored in water in T-Plant Canyon Pool Cell 4.

- Fast Flux Test Facility (FFTF) SNF - stainless steel-clad fuel stored in liquid sodium at the FFTF, consisting mostly of plutonium and uranium oxide fuel, but also uranium and/or plutonium metals, and carbide and nitride fuel.

- Miscellaneous commercial and experimental SNF - consisting mainly of Zircaloy-clad uranium dioxide fuel stored in air in the 324, 325, and 327 buildings; TRIGA (training, research, and isotope reactors built by General Atomics) fuel stored in water in the 308 Building; miscellaneous fuel stored in air-filled shielded containers at the 200-West Area burial grounds; and aluminum-clad, uranium-aluminum alloy fuel stored in air in the Plutonium Finishing Plant.

Plans for management of Hanford SNF are included in the *Hanford Spent Nuclear Fuel Project, Recommended Path Forward* (Fulton 1994) and the *Spent Nuclear Fuel Project Technical Baseline Document Fiscal Year 1995* (WHC 1995). It should be noted, however, that the SNF management program has continued to evolve since these documents were issued or drafted. Similarly, Hanford site-specific environmental documentation that will be required to support the Hanford SNF management program continues to evolve. Spent nuclear fuel EISs that are being prepared or that will be prepared include this programmatic EIS and a Hanford site- specific K Basins EIS. The programmatic EIS will lead to a record of decision that is scheduled to be published in June 1995. That record of decision will specify what SNF will be managed at which DOE sites, Naval Reactor Propulsion Program sites, or other sites. The K Basins EIS is expected to result in a record of decision that specifies where and how to relocate, stabilize, and safely store N Reactor and single-pass reactor SNF from the K Basins to address the urgent need to remedy safety and environmental vulnerabilities. The K Basins EIS record of decision will address management of this SNF over a 40-year period or until ultimate disposition.

During negotiations on the Fourth Amendment to the Tri-Party Agreement (TPA), the DOE, the State of Washington Department of Ecology, and the EPA agreed to an enforceable milestone that indirectly required issuing that record of decision by June 1996. The record of decision on the K Basins EIS would be dependent on the programmatic EIS record of decision. Other environmental documentation (EAs or EISs) will be prepared for any proposed actions related to SNF that are not specifically covered in the programmatic EIS or in the K Basins EIS.

Assuming the EISs are prepared as planned, the Hanford SNF management plan would identify and implement management approaches that will provide safe, cost-effective storage of SNF at existing facilities. Activities to identify, and then implement, the SNF management approach follow:

- Issuing the records of decision that are expected to result from the programmatic EIS and the K Basin EIS.
- Achieving accord with the TPA or renegotiating activities and milestones, as necessary.
- Providing facilities for SNF management as necessary to implement the EIS records of decision. SNF remaining onsite, as a result of the programmatic EIS record of decision could be placed in wet or dry storage in the 200-East Area until a decision on ultimate disposition has been made.
- Identifying and developing pathways for ultimate disposition of the SNF.
- Providing facilities and systems for preparing SNF for ultimate disposition. N Reactor and single-pass reactor SNF would be stabilized, as necessary, to implement the K Basins EIS record of decision. It is possible this stabilized form would be a metal or an oxide. Suitability of other SNF for ultimate disposition in its current form is yet to be demonstrated, but it is possible that FFTF and Shippingport SNF may not require further stabilization.

While the SNF management approach is being defined, the following key, near-term actions at the existing facilities are being implemented or are planned:

- Upgrading water treatment systems and retrieving sludges from the basins' floors.
- Performing necessary safety and security upgrades (e.g., water systems) to extend facility life until SNF removal can be accomplished.
- Transferring SNF from liquid-sodium storage at the FFTF to dry storage in interim storage casks. This activity would be integrated with FFTF deactivation.

- Transferring small quantities of SNF between existing facilities where deemed necessary to comply with other Hanford requirements.

Discussion of the SNF inventory and plans for managing that inventory are provided in the following sections. Planned SNF management activities are summarized in Table 2-1. Additional details on existing storage facilities are in Chapter 3.

2.3.1 N Reactor Spent Nuclear Fuel

N Reactor SNF is stored in three facilities (Bergsman 1994):

- 952 metric tons of uranium in 3815 closed canisters in the 105-KW Basin. The water in this basin has only low levels of radionuclide contamination.
- 1144 metric tons of uranium in 3666 open canisters in the 105-KE Basin. The water in this basin is contaminated with radionuclides, and there is a thick layer of sludge on the basin floor.
- 0.3 metric tons of uranium in the form of intact Mark IV fuel elements and fuel element pieces stored in air on the floor of PUREX dissolver cells A, B, and C.

Until recently, plans included 1) containerizing the fuel and sludge stored in the 105-KE Basin into Mark II (sealed) canisters; and 2) transferring the spent fuel in PUREX to the 105-KE Basin and containerizing it in the basin. Alternative approaches to each of these plans, including alternative containerization of fuel and sludge at the 105-KE Basin, expedited fuel removal from the K Basins and dry storage of fuel at PUREX, have been evaluated, and a path forward for these materials selected. PUREX SNF would be transferred to the K Basins and subsequently managed with the existing K Basins SNF inventory pending issuance of an environmental assessment.

Expedited fuel removal from the K Basins has been selected in lieu of containerization because of benefits to worker safety and/or the environment. The 105-K Basins SNF would be relocated to a storage facility in the 200 Area, pending completion of the K Basins EIS. The impacts associated with implementation of this path forward are within the envelope of impacts analyzed in this EIS.

Table 2-1. Summary of planned spent nuclear fuel management activities.[a]

Spent nuclear fuel	Activity	Schedule	Status
N Reactor	Transfer SNF stored at PUREX to 105-K Basins	Complete by 1/96	Environmental assessment submitted
	Remove SNF and sludge from 105-K Basins per DNFSB Recommendation 94-1; transfer onsite to storage system.	Complete by 12/99	K Basins EIS initiated
Fast Flux Test Facility	Transfer SNF from liquid sodium to dry storage	Deliver first 10 casks by 8/95	Environmental assessment submitted
	Transfer small quantities SNF onsite to satisfy physical security requirements.	10/98	Environmental assessment submitted
Single-pass reactor	Transfer SNF stored at PUREX to 105-K Basins	Complete by 1/96	Environmental assessment submitted
Shippingport Core II	Transfer SNF from T-Plant onsite		Plans will be developed pending ROD for this EIS
Miscellaneous in 300 Area	Transfer SNF from 324/325/327 buildings onsite	Complete in mid-1999	Environmental assessment planned and will be prepared pending ROD for the EIS
	Transfer TRIGA SNF from 308 Building onsite	Complete in 1996	Environmental assessment submitted
Miscellaneous in 200 Area	May be transferred onsite		

a. Source: Bergsman (1995).

In addition, work is ongoing to characterize the N Reactor and single-pass reactor fuel to provide data relevant to assuring continued safe storage and developing plans for future actions. Recent commitments to the Defense Nuclear Facilities Safety Board have set a date of December 1999 for completing removal of the SNF from the 105-K Basins.

Other N Reactor SNF, which may be recovered as a result of N Basin deactivation, would also be transferred to the 105-K Basins. A small quantity of this material (less than 0.5 MTHM) in the form of fuel fragments and chips is suspected to be in the sludge at the bottom of N Basin.

2.3.2 Single-Pass Reactor Spent Nuclear Fuel

The single-pass reactor SNF consists of residual fuel elements from the 105-KW and 105-KE reactors, plus residual elements from the clean-out of the 105-C and 105-D storage basins. Currently, 138 elements [0.4 metric tons of uranium (MTU)] are stored in the 105-KE Basin and 47 elements (0.1) are stored in the 105-KW Basin. In addition, four buckets filled with 779 single-pass reactor fuel elements are stored in the PUREX storage basin.

It was planned that the single-pass reactor fuel stored in PUREX would be transferred to the 105-KE Basin, containerized, and possibly transferred to the 105-KW Basin before the previously planned Hanford SNF EIS record of decision would be issued. Activities to implement this action were initiated (Bergsman 1995). In parallel, alternative dry storage of this fuel was considered, consistent with the dry storage evaluation for N Reactor fuel at PUREX. To enable expeditious deactivation of the PUREX plant in support of the Hanford Site cleanup mission and because of the minimal impacts associated with relocation of this SNF to the 105-K Basins, shipment to the 105-K Basins was selected as the preferred approach for managing this SNF until issuance and implementation of the K Basins EIS record of decision. The SNF may be shipped directly to the 105-KW Basin instead of the 105-KE Basin and would be stored in a manner consistent with the requirements of the selected storage basin. The impacts associated with implementation of this path forward are within the envelope of impacts analyzed in this EIS.

2.3.3 Fast Flux Test Facility Spent Nuclear Fuel

The SNF from FFTF is stored in the following four FFTF locations, all of which use liquid sodium for cooling:

- the reactor core with a capacity of approximately[a] 82 fuel assemblies
- in-vessel storage with a capacity of 54 fuel assemblies
- interim decay storage with a capacity of 112 fuel assemblies and a limitation of 10 kilowatts per assembly
- the Fuel Storage Facility with a capacity of 380 fuel assemblies[b] and a limitation of 1.4 kilowatts per assembly.

The 1993 inventory of irradiated SNF at FFTF consists of fuel from 329 assemblies; an additional 55 non-irradiated driver fuel assemblies exist. Some irradiated fuel assemblies have been disassembled, with the fuel now placed in 40 Ident 69 containers or in the Interim Examination and Maintenance Cell. Some irradiated fuel has been shipped offsite, but is expected to be returned to Hanford.

The DOE plans to transfer FFTF spent nuclear fuel from the liquid sodium-cooled storage facilities into dry storage casks. These interim storage casks would hold six or seven assemblies per cask. Delivery of an initial ten casks has been scheduled for August 1995 and an environmental assessment for this activity has been submitted (Bergsman 1995). The majority of the casks would be sited in the 400 Area; however, a few may be sited at the Plutonium Finishing Plant because of requirements for additional physical security. A small fraction of the FFTF SNF is sodium bonded, and may be shipped directly offsite without emplacement in dry storage casks if the decision in this EIS is to relocate these materials to another DOE site.

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a. Capacity for each core-loading varies.

b. The Fuel Storage Facility actually has a capacity of 466 fuel assemblies, but is limited to only 380 because of criticality requirements.

2.3.4 Shippingport Core II Spent Nuclear Fuel

The Shippingport Core II spent nuclear fuel is stored in water in the 221-T Building (T-Plant) Canyon Pool Cell 4. The 72 standard blanket assemblies will remain in basin storage in T-Plant until site-specific NEPA review is completed to enable implementation of dry storage or transfer offsite. Site-specific NEPA review will not be initiated until issuance of the record of decision for this EIS. (One un-irradiated blanket assembly is also stored in air in the T-Plant.)

2.3.5 Miscellaneous Spent Nuclear Fuel

A variety of miscellaneous spent nuclear fuel is stored in the 300 Area, Plutonium Finishing Plant, and low-level burial grounds (Bergsman 1994). Specific actions that have been identified (Bergsman 1995) follow:

- The spent nuclear fuel stored in air in the 324, 325, and 327 buildings (mostly commercial, light-water reactor fuel, i.e., Zircaloy-clad uranium dioxide) is planned for relocation onsite; an environmental assessment for this activity will be prepared. The planned storage facility is a dry storage cask.

- TRIGA fuel stored in water in the 308 Building is planned for relocation onsite to the 400 Area so that the 308 Building can be deactivated; an environmental assessment has been submitted for this activity. Alternative disposition of the TRIGA fuel may be implemented; transfer of this fuel to the Idaho National Engineering Laboratory (INEL) is assumed in the INEL 1992/1993 Planning Basis Alternative.

- Miscellaneous fuel residues in the 200 Area are currently being managed as remote-handled transuranic waste. The TRIGA SNF at the burial grounds will be relocated onsite during burial grounds retrieval operations.

3. SPENT NUCLEAR FUEL MANAGEMENT ALTERNATIVES

3.1 Description of Alternatives

Five major alternatives are being evaluated for safely storing SNF until ultimate disposition is determined. These five alternatives are 1) No Action, 2) Decentralization (with a subset of local stabilization and storage options), 3) 1992/1993 Planning Basis, 4) Regionalization (with options A, B1, B2, and C), and 5) Centralization (minimum and maximum options). The five alternatives and their impacts are being evaluated concurrently by the sites or agencies potentially affected by these alternatives, including Hanford, Savannah River Site (SRS), Idaho National Engineering Laboratory (INEL), Oak Ridge National Laboratory (ORNL), the Nevada Test Site (NTS), and the Naval Nuclear Propulsion Program.

This chapter describes the spent fuel inventories, activities, and facilities anticipated at Hanford under the various storage alternatives. The inventory of SNF expected to be stored at Hanford under each alternative is summarized in Table 3-1. There are eight types of fuel listed in Table 3-1 to represent the wide variety of SNF currently held at various sites across the United States. In addition, the United States has obligations for some SNF held in foreign countries. The specific kinds of SNF held at Hanford that contribute toward the total SNF inventory are shown in parentheses in column one of Table 3-1. In terms of metric tons of heavy metal, Hanford has about 80 percent of DOE's current SNF inventory, primarily because of the large inventory of spent fuel remaining from the shut-down N Reactor. The Centralization Alternative minimum option is not shown in Table 3-1 because the inventory would eventually be zero at Hanford under this option, as it is in the Regionalization Alternative Option C. An overview of the SNF inventory as of the year 2035, planned activities, and existing and new facilities that may result under each of the five storage alternatives is provided below.

The No Action Alternative described in Subsection 3.1.1 forms the basis for comparison with the remaining four storage alternatives and includes descriptions of the expected activities, and existing storage facilities. Decentralization (Subsection 3.1.2), the 1992/93 Planning Basis (Subsection 3.1.3), Regionalization (Subsection 3.1.4), and Centralization (Subsection 3.1.5) are discussed in the remaining sections.

Table 3-1. Spent nuclear fuel inventory at Hanford under the various storage options as of 2035 in MTHM.[a,b]

Fuel type (name of Hanford SNF that is part of this type)	No Action and Decentralization	1992/1993 Planning Basis	Regionalization A[c]	Regionalization B1[d]	Regionalization B2[e]	Regionalization C[f] and Centralization minimum option	Centralization maximum option
Naval SNF	0.00	0.00	0.00	10.23	65.23	0.00	65.23
Savannah River and aluminum-clad	0.00	0.00	0.00	8.76	8.76	0.00	213.09
Hanford (N Reactor and single-pass reactors)	2103.17[g]	2103.17	2103.17	2103.17	2103.17	0.00	2103.17
Graphite	0.00	0.00	0.00	27.60	27.60	0.00	27.61
Commercial miscellaneous fuels	2.30	2.30	0.00	125.18	125.18	0.00	156.51
Experimental, stainless steel clad (FFTF)	11.27	11.23	0.00	90.12	90.12	0.00	96.51
Experimental, Zircaloy clad (Shippingport)	15.70	15.70	0.00	64.84	64.84	0.00	77.99
Experimental, other such as ceramic, liquid/salt, etc.	0.00	0.00	0.00	0.29	0.29	0.00	1.70
TOTALS:	2132.44	2132.40	2103.17	2430.19	2485.19	0.00	2741.80

a. MTHM - Metric tons of heavy metal (thorium, uranium, and plutonium as applicable).

b. Source: Wichmann (1995). Quantities of SNF within a given category may be the result of adding together several quantities, some large and some small, stored at different locations. Individual values are known to within about 1%. Additional digits are shown in the table as a check on calculations, but inventory totals are known to only two significant figures.

c. All Hanford production SNF remains at Hanford. All other SNF goes to INEL (including Hanford commercial, experimental stainless-steel-clad, and TRIGA).

d. All SNF currently located or to be generated in the U.S. west of the Mississippi River is sent to and stored at the Hanford Site, with the exception of Naval SNF.

e. All SNF currently located or to be generated in the U.S. west of the Mississippi River and all Naval SNF are sent to and stored at the Hanford Site.

f. All Hanford Site SNF and all other SNF currently located or to be generated in the U.S. west of the Mississippi River is sent to and stored at either INEL or NTS. For Hanford, this alternative is identical to the Centralization Alternative minimum option (SNF is shipped offsite).

g. This represents the post-irradiation (end-of-life) quantity. The pre-irradiation quantity, (2116.67 MTHM) is sometimes quoted.

3.1.1 No Action Alternative

Under the No Action Alternative, only those actions that are deemed necessary for continued safe and secure management of the SNF would be conducted. Thus, the existing SNF would be maintained close to its current storage locations, and there would be minimal facility upgrades. Activities required to store SNF safely would continue at each specific site (DOE 1993b).

A description of the anticipated activities that would be necessary under the No Action Alternative is provided in Subsection 3.1.1.1, followed by descriptions of existing facilities (Subsection 3.1.1.2), and any new facilities (Subsection 3.1.1.3). A comprehensive inventory and description of the fuel at Hanford as of January 1993 is given by Bergsman (1994). That report provides detailed information on many of the spent fuel designs and radionuclide inventories.

3.1.1.1 Anticipated Activities. In order to carry out the No Action Alternative, the following activities would occur at the Hanford Site:

- Characterization of the defense production reactor fuel would proceed to establish the basis for safe storage.
- Fuel and sludge would be containerized at the 105-KE Basin or other onsite location.
- The first 10 dry storage casks would be procured for Fast Flux Test Facility (FFTF) fuel.

Consolidation of SNF from defense production reactors into the 105-KW Basin could occur. Other fuel may be transferred to dry cask storage where required for safety.

3.1.1.2 Description of Existing Facilities. SNF is presently located in 11 facilities on the Hanford Site: 105-KE and 105-KW Basins at the north end of Hanford in the 100-K Area; T Plant, low-level waste burial grounds, and Plutonium Finishing Plant in the 200 West Area; Plutonium and Uranium Recovery through EXtraction (PUREX) plant in the 200 East Area; FFTF in the 400 Area; and 308, 324, 325, and 327 buildings in the 300 Area in the southeast corner of the site. Continued storage in these facilities is being evaluated because the No Action Alternative includes activities required to ensure safe and secure storage. The Plutonium

Finishing Plant and PUREX facilities are excluded from this evaluation because SNF will not remain in those two facilities under any of the alternatives. For the purposes of this analysis, SNF at PUREX is assumed to be relocated to the K Basins.

Most of the facilities at the Hanford Site are decades old, some over 40 years, except for the FFTF and its associated storage buildings. A general description, the capacity for additional storage of SNF, and the means by which SNF can be received or removed from each facility are provided in Table 3-2. The dimensional information is for the actual storage area and not for the entire facility in order to provide a basic idea of the storage area required for that specific inventory of SNF. In many cases, such as the facilities in the 300 Area, only small portions of the actual facilities are used to store the spent fuel.

The K Basins contain the vast majority of the SNF at Hanford. The T-Plant, 308, 325, and 327 buildings, and the Plutonium Finishing Plant contain small amounts of stored SNF of various kinds. Four FFTF locations contain all the FFTF spent fuel, presently stored in sodium: the Reactor Core, In Vessel Storage, Interim Decay Storage, and Fuel Storage Facility (a building separate from the reactor containment building). The first of 60 new dry storage casks are expected to be available for FFTF fuel by late 1995. The existing facilities have very little additional capacity (see Table 3-2). While there is presently excess capacity in the K Basins, this is expected to be consumed by the planned operations, regardless of the storage alternative chosen.

The accessibility and limits on loading SNF are provided as key factors in movement of any fuel from these facilities to other locations on or offsite. Rail access is available at the facilities storing most of the fuel (K Basins, PUREX, and T Plant); truck shipments would be used for the rest. Acceptable casks and procedures for moving these casks may require evaluation in many cases. Additional details on these facilities are provided by Bergsman (1994), Bergsman (1995), and Monthey (1993).

The changes to the existing facilities that were analyzed under the No Action Alternative of SNF storage are shown in Table 3-3.

Table 3-2. Description of existing facilities (Bergsman 1994; Bergsman 1995).

Facility	Description	Capacity	Access
105-KE Basin	Water storage pool; 38 m x 20 m x 6 m deep; concrete walls and floor; no sealant or liner	75% full, 100% full after containerization	By rail 27 MT crane, fairly restrictive
105-KW Basin	Water storage pool; 38 m x 20 m x 6 m deep; concrete walls and floor; epoxy sealant; no liner	75% full[a]	By rail 27 MT crane, fairly restrictive
T Plant: Cell 4	Water storage pool; 4 m x 8.4 m x 5.8 m deep (water)	50% full	By rail or truck All fuel handling remote
PUREX Plant: East end of 202A Bldg. plus Dissolver Cells A, B, and C	Water storage pool; 9.5 m x 6.1 m x 5.2 m deep; Dissolver Cell sizes vary	No additional capacity	Shipment by rail 36 MT crane
Plutonium Finishing Plant: 2736-ZB Bldg.	Dry storage in 55 gal drum	No additional capacity	Shipment by truck
Fast Flux Test Facility: Reactor in-vessel storage, interim decay storage, and fuel storage facility storage locations	Liquid sodium pool storage (fuel storage facility is separate from reactor containment building, with limit of < 1.4kW/assembly)	More than 75% full	By truck 91 MT Crane
200 Area LL Burial Grounds: 218-W-4C Trench 1 and 7; and 218-W-3A Trench 8 and S6	Dry, retrievable storage; 13 lead-lined, concrete-filled 208 liter drums, soil covered; 22 concrete casks (1.66 m x 1.66 m x 1.22 m or 1.92 m high), soil covered; 39 EBR II casks (1.5 m high x 0.4 m diameter), soil covered; 1 Zircaloy Hull Container (152 cm long x 76 cm diameter)	Large additional capacity	By truck
308 Building Annex: Neutron Radiography Facility	Built in late 1970's water storage pool; 2.8 m diameter x 6 m deep	Small additional capacity	Truck shipments 4.5 MT crane
324 Building: B and D Cells	Dry storage in air; B Cell: 6.7 m x 7.6 m x 9.3 m high (SNF uses < 10% of floor space). D Cell: 4 x 6.4 m x 5.2 m high (small part for fuel), thick concrete walls and floors with steel liners	Small additional capacity	Truck shipments only B Cell - 2.7 and 5.4 MT cranes; Airlock - 27 MT crane
325 Building: A and B Cells in 325 Radiochemical Facility; 325 Shielded Analytical Laboratory	Dry storage in air 325A - 1.8 m x 2.1 m x 4.6 m high (typical cell) 325B - 1.7 m x 1.7 m floor area (typical cell)	Small additional capacity	Truck shipments only 325A - 27 MT crane 325B - 2.7 MT crane
327 Building: A - F and I Cells; Upper and Lower SERF; Dry Storage vault; EBR II cask; Large Basin	Dry storage in air, except for water in large basin; variety of cell sizes, but storage only for fuel research	Small additional capacity	No direct rail Truck shipments 13.5 and 18 MT cranes

a. If 105-KE Basin fuel is consolidated with 105-KW Basin fuel, 105-KE Basin would be shut down. The storage capacity of 105-KW Basin would be increased by replacing all the storage racks to allow multitiered stacking of fuel storage canisters and by making minor facility modifications.

Table 3-3. Assumed changes to existing Hanford facilities in the No Action Alternative.

Facility	Facility changes
105-KE Basin	Fuel and sludge to be containerized; plans to upgrade safety and security systems
105-KW Basin	Fuel is already containerized; plans to upgrade safety and security systems
T Plant	None
PUREX Plant	Fuel to be moved to alternative location (assumed to be 105-K Basins for this alternative)
Plutonium Finishing Plant	None
Fast Flux Test Facility	None: Procure 10 dry storage casks by 8/95 (Bergsman 1995). Casks to weigh 50 T with storage cavity 3.8 m high x 0.56 m diameter (Bergsman 1994)
200 Area LL Waste Burial Grounds	None
308 Building Annex	None
324 Building	None
325 Building	None
327 Building	None

3.1.1.3 Description of New Facilities. No new buildings were analyzed for the Hanford Site under the No Action Alternative. The only activities that were analyzed are those described for containerizing the N Reactor fuel and procuring casks for storage of FFTF fuel. The casks would be stored above ground on an existing concrete pad at the FFTF (Bergsman 1995). Major changes in rail, electrical, water, or other utilities are not expected under this alternative.

3.1.2 Decentralization Alternative

In the Decentralization Storage Alternative, as in the No Action Alternative, the current spent fuel inventory would continue to remain close to the point of generation or defueling. There are some existing storage sites that may receive or ship spent fuels, such as naval spent fuel, under one of several options under the Decentralization Alternative, but these options do not impact Hanford (DOE 1993a). No SNF would be shipped offsite or received from other storage locations outside of Hanford, but local transport might take place to support safety requirements and research and development. The Decentralization Alternative differs from the No Action Alternative in that significant facility development and upgrades are assumed, and spent fuel characterization, research and development, and possibly stabilization would occur.

Summaries of the anticipated activities (Subsection 3.1.2.1) and facility requirements (Subsections 3.1.2.2 and 3.1.2.3) are provided below.

3.1.2.1 Anticipated Activities. The Decentralization Alternative would include the three activities (fuel characterization, fuel and sludge containerization, and cask procurement for FFTF fuel) mentioned above in Subsection 3.1.1 for the No Action Alternative as well as the following general activities:

- Characterization of defense production fuels (N Reactor and single-pass reactor) to determine the feasibility of dry storage
- Evaluation of dry storage for other fuels (Shippingport Core II, FFTF, miscellaneous)
- Research and development on N Reactor fuel stabilization
- Construction and utilization of wet and/or dry storage facilities as well as a stabilization facility to support storage.

Only the defense fuels are being considered for wet storage, but dry storage in casks or vaults could be used for all or part of Hanford's spent fuel inventory under various options (Bergsman 1995). There are four basic options considered for storage of the spent fuels at Hanford under the Decentralization Alternative. Options W and X include both wet and dry storage: wet storage for defense fuels and dry storage for all other spent fuels in either a vault or casks. Options Y and Z involve only dry storage, again either in a vault or casks, but these options include one of three stabilization options for the metallic defense fuels.

The three potential processes considered for stabilizing the defense fuels in conjunction with Options Y and Z are shear/leach/calcine (P), shear/leach/solvent extraction (Q), and drying and passivation (D). Process P consists of shearing the fuel into a continuous dissolver and dissolving it in a nitric acid solution. Eventually, the processed material (without any radionuclide removal) is calcined, pressed into a ceramic waste form, and sealed in metal canisters.

Process Q uses solvent extraction by which metallic defense fuels are dissolved, separating uranium and plutonium and a liquid high-level waste stream that would most likely be vitrified for disposal in a geologic repository. In Process Q it is assumed that the process would be carried out on the Hanford Site. In commenting on the draft EIS, British Nuclear Fuels Limited (BNFL) proposed such processing be carried out in their facilities overseas. A discussion of the proposed sub-option is provided in Attachment B. Except for the additional impacts associated with transporting SNF from the Hanford Site to a West Coast shipping port, transoceanic shipment, transport of the SNF overland to BNFL facilities, and return shipment of resource materials (uranium-trioxide and plutonium-dioxide) and vitrified high-level waste, environmental impacts would be similar to those determined for Process Q.

Process D consists of drying and passivating the spent fuel and then canning it for storage. The relationships between the storage and stabilizing options are shown in Table 3-4.

Option W involves moving the N Reactor fuel from the existing basin storage into a new basin to be built by the year 2001. Simultaneously, a modular dry vault would be built for storage of the rest of the spent fuel at Hanford. Option X considers the use of casks for dry storage instead of the vault, but still requires moving the N Reactor fuel to a new basin. The casks would be placed on concrete pads outside of any buildings and would include two types of cask designs: concrete modules holding a storage cask, and upright concrete casks designed specifically for the FFTF fuel. Option Y would result in all of the non-defense spent fuel at Hanford being placed in a large vault facility. The defense fuel would require processing in a new facility by one of three options (P, Q, or D) prior to canning and placement in storage. The defense fuels processed using Option P or Option D would be stored in the vault; however, Option Q would result in several products that would be stored or processed further as high-level waste (Bergsman 1995). The final option, Option Z, is similar to Option Y except that casks would be used instead of a dry storage vault for all of the nondefense spent fuels. The defense fuels are handled as in Option Y. Additional details are provided by Bergsman (1995).

Table 3-4. Options under the Decentralization Alternative for Hanford.

Storage option	Stabilization option	Description	Facility requirements
W	None	Wet storage of defense fuels Dry storage of other fuels	New basin New vault
X	None	Wet storage of defense fuels Dry storage of other fuels	New basin New casks
Y	P, Q, or D	Dry storage of all fuel; stabilize defense fuels prior to storage	New vault; new processing facility [calcining (P), solvent extraction (Q), or drying and passivation (D)]
Z	P, Q, or D	Dry storage of all fuel; stabilize defense fuels prior to storage	New dry storage casks; new processing facility [calcining (P), solvent extraction (Q), or drying and passivation (D)]

3.1.2.2 Description of Existing Facilities and Impacts from the Decentralization Alternative. The description of the existing facilities used to store SNF at Hanford was provided in Subsection 3.1.1.2. The Decentralization Alternative would impact the facilities beyond that already mentioned for the No Action Alternative to the extent that fuel would be removed from several of them: the Shippingport fuel would be removed from T Plant to a designated interim storage location on site; FFTF fuel would continue to be removed from the sodium-cooled storage facilities and placed in dry storage casks; and fuel in the 200-W burial grounds might be relocated onsite.

As shown in Table 3-2, there is very little excess capacity in any of the facilities in which fuel is currently stored. The storage basins, in addition to being old, were built for temporary holding, for a matter of months only; hence, bringing them up to standards for prolonged storage would be fraught with problems and would not be cost-effective. Except for the burial grounds, the locations in which SNF is currently held in air were not intended for prolonged storage either, having been built for temporary holding for research and development or pre-processing. The FFTF storage facilities are all dependent on maintaining sodium in the liquid state as coolant and storage medium, which is not cost-effective for 40 years of storage for nonbeneficial use. Hence, the existing facilities are not considered for use in the 40 year storage scenario.

3.1.2.3 Description of New Facilities. A minimum of two new facilities are required, regardless of which option is chosen for storing spent fuel under the Decentralization

Alternative. Both Options W and X require a new basin and either a new vault or a new cask storage facility. Descriptions of these potential new facilities are provided in Table 3-5. A proposed site consisting of about 260 hectares (one-quarter section) for construction of all new facilities is located as shown in Figure 4-1. The cask facility would cover about twice as much land area as a vault facility and would involve modular systems placed outside on concrete pads. While the basin requirement is dropped for Options Y and Z, a process facility is needed for the metallic defense fuels in addition to the new dry storage facility. The specifics of this facility vary depending on whether they involve shear/leach/calcining (process P), shear/leach/solvent extraction (process Q), or drying and passivation (process D). For process Q, it is assumed that a vitrification plant and storage facilities will be available for the processed spent fuel that would then consist of three products. The vitrification plant and storage for high-level wastes are part of the overall plan for Hanford.

The potential processing facilities that will result from this alternative will require increased utilities, compared with the new dry storage facilities that are not expected to have major utility requirements. A rail system for receiving spent fuel at the various facilities may be required and could be tied into the existing system. Water requirements are expected to be insignificant. Estimates of the power requirements for processes P, Q, and D are 10 megawatts, 18 megawatts, and 3 megawatts, respectively. While the existing excess electrical capacity of 21 megawatts would be sufficient for one of these facilities, other potential uses of the existing electrical power capacity may require upgrading the existing power system (Bergsman 1995).

3.1.3 1992/1993 Planning Basis Alternative

The 1992/1993 Planning Basis Alternative defines those activities that were already scheduled at the various sites for the transportation, receipt, processing, and storage of SNF.

3.1.3.1 Description of Spent Fuel Inventory As in the previous two alternatives, no new spent fuel would be received at Hanford under the 1992/1993 Planning Basis Alternative. However, the 101 spent fuel elements currently in the 308 Building from TRIGA reactors and the small amount of TRIGA fuel from Oregon State University currently in the 200-W Area burial grounds would be shipped to INEL.

Table 3-5. Description of required facilities under the Decentralization Alternative.[a]

New facility	Description	Capacity
Water Basin (W, X)	**Building:** 110 m long x 42.7 m wide x 19.8 m high **Land use:** < 8094 m^2 (< 2 acres) **Water storage pool:** rectangular, 520 m^2, cast-in-place concrete **Canisters:** double barreled, each 0.23 m diameter x 0.74 m high **Construction:** 3 year duration, operation by 2001	2103 MTU in 8000 canisters
Dry Storage Vault Facility (W)	**Building:** 39.6 m long x 48.8 m wide x 19.8 m high **Land use:** <4047 m^2 (<1 acre) **Modular vault:** metal tubes vertically arrayed in cast-in-place concrete structure; inert cover gas; natural convection cooling. **Canisters:** short, 0.508 m diameter x 3.96 m (FFTF fuels); long, 0.559 m diameter x 4.57 m (other non-defense fuels) **Construction:** 3 year duration, operation by 2001	30 MTHM in 60 short and 25 long canisters
Dry Storage Cask Facility (X)	**Building:** none, concrete pads **Land use:** <8094 m^2 (<2 acres) **Cask Systems:** 1) FFTF casks, 2.29 m diameter x 4.57 m high, 45.4 MT each, 2) Concrete module with fuel cask; reference storage module is 2.96 m wide x 5.52 m deep x 4.57 m high **Canisters:** 0.508 m diameter x 3.96 m (FFTF cask); 1.68 m diameter x 4.88 m long, weighs 90.8 MT (storage module) **Construction:** 3 year duration, operation by 2001	30 MTHM, 60 cask/ canisters (FFTF design) and 6 storage modules/ casks
Shear/Leach/Calcine Process or Z Facility (Y)	**Building:** multilevel, steel-reinforced, cast in place concrete; 110.3 m long x 55.2 m wide x 25.9 m high (15.8 m above grade); shielded main canyon is 6.1 m wide x 70.1 m long x 25.9 m high; **Land Use:** 6070 m^2 (1.5 acres) **Operation:** 24 hours/day, 7 days/week for 4 years to stabilize defense fuels; 75% efficiency; 280 day/year **Construction:** 3 year duration, operation by 2001	2103 MTU in 4 years 2.5 MTU/day
Dry Storage Vault Facility (Y)	**Building:** 100.6 m long x 88.4 m wide x 18.3 m high **Land use:** < 8094 m^2 (< 2 acre) **Modular vault:** metal tubes vertically arrayed in cast-in-place concrete structure; inert storage atmosphere; natural convection cooling. **Canisters:** 0.559 m diameter x 4.11 m (defense fuels); short, 0.508 m diameter x 3.96 m (FFTF fuels); long, 0.559 m diameter x 4.57 m (other non-defense fuels) **Construction:** 3 year duration, operation by 2001	2133 MTHM in ~1200 defense canisters, 60 short and 25 long non-defense canisters
Dry Storage Cask Facility (Z)	Same as Dry Cask Storage Facility described for Option X **Land use:** 20,234 m^2 (5 acres) **Canisters:** add storage modules/casks for stabilized defense fuels; same storage container dimensions as for Option X	2133 MTHM in 60 cask/ canisters (FFTF), 230 modules/casks (defense), and 6 modules/ casks (other non-defense)
Solvent Extraction Fuel Process Facility (Y or Z)	**Building:** multilevel, steel-reinforced, cast in place concrete; 26.5 m long x 77.7 m wide x 25.9 m high (15.8 m above grade); shielded main canyon is 6.1 m wide x 76.2 m long x 25.9 m high; **Land Use:** 6070 m^2 (1.5 acres) **Canisters:** generates 2 kg/MTU of fuel processed, resulting in about 30 cans of glass for 2103 MTU of fuel **Operation:** 24 hours/day, 7 days/week for 4 years to stabilize defense fuels; 75% efficiency; 280 day/year **Construction:** 3 year duration, operation by 2001	2103 MTU in 4 years 2.5 MTU/day
Fuel Drying and Passivation Facility (Y or Z)	**Building:** multilevel, steel-reinforced, cast in place concrete; 115.8 m long x 64.0 m wide x 25.9 m high (15.8 m above grade); shielded main canyon is 6.1 m wide x 54.9 m long x 25.9 m high; **Land Use:** 6070 m^2 (1.5 acres) **Operation:** 24 hours/day, 7 days/week for 4 years to stabilize defense fuels; 75% efficiency; 280 day/year **Construction:** 3 year duration, operation by 2000	2103 MTU in 4 years, 2.5 MTU/day

a. Source: Bergsman (1995).

3.1.3.2 Anticipated Activities Most of the activities previously discussed for the decentralization storage alternative were already planned prior to this review. It was expected that all newly generated SNF that was owned by the U.S. Government would be sent to either INEL or to SRS. No new spent fuel was expected to be shipped to Hanford other than possibly limited quantities of material for research or other scientific endeavors supporting the nuclear industry. Upgrades and replacements of existing storage capacity were already planned and would involve those facilities described in Subsection 3.1.2 for the Decentralization Alternative. Thus, the activities that would be conducted under the 1992/1993 Planning Basis are the same as for the Decentralization Alternative under the four options listed in Table 3-4, except for the additional activity of shipping TRIGA spent fuel to INEL.

3.1.3.3 Description of Existing Facilities and Changes Required by Alternative The description provided in Subsection 3.1.1.2 on the existing facilities for storing SNF at Hanford also applies to this alternative. No additional changes to facilities are anticipated from the 1992/1993 Planning Basis except that the 308 Building and the 200W Area burial grounds would no longer contain TRIGA spent fuel.

3.1.3.4 Description of New Facilities. The facilities that would be required under the 1992/1993 Planning Basis are the same as those shown previously in Table 3-5 for the Decentralization Alternative. The impact on existing utilities would be the same as for the Decentralization Alternative, namely from 3 to 18 megawatts of power for stabilization facilities and minimal other impacts.

3.1.4 Regionalization Alternative

This alternative provides for the redistribution of SNF to candidate sites based on similarity of fuel types (Option A) or on geographic location (Options B1, B2, and C), in order to optimize the storage of SNF owned by the U.S. Government.

The Regionalization Alternative as it applies to the Hanford Site consists of the following options:

- Option A (regionalized by fuel type) - Defense production SNF would remain at Hanford; other types of SNF would be sent to INEL.

- Option B1 (geographic regionalization) - All SNF west of the Mississippi River except Naval SNF would be sent to Hanford.

- Option B2 (geographic regionalization) - All SNF west of the Mississippi River and Naval SNF would be sent to Hanford.

- Option C (geographic regionalization) - All Hanford SNF would be sent to INEL or NTS.

Facilities and features of Regionalization Option A would be the same as those described for Hanford defense production fuel in the Decentralization Alternative. The facilities and features for all other Hanford SNF would be very similar to those described for that SNF in the Centralization Alternative minimum option.

Facilities and features of Regionalization Options B1 and B2 would be incremental to those described for the Decentralization Alternative and would include facilities and features similar to those described in the Centralization Alternative maximum option.

Facilities and features of Regionalization Option C would be equivalent to those described for the Centralization Alternative minimum option.

3.1.4.1 Description of Spent Fuel Inventory. The spent fuel inventory that would be stabilized and/or stored for each of the Regionalization options is shown in Table 3-1.

3.1.4.2 Activities Required by Each Option.

<u>Option A, Suboption X</u>

- wet storage of N Reactor and single-pass reactor fuel

- shipment of other Hanford Site fuel to INEL

- use of existing facilities (FFTF and T Plant) and new wet pool facilities to load shipping casks.

For N Reactor and single-pass reactor fuel, this option is the same as the Decentralization Alternative; for all other Hanford Site fuel, this option is nearly the same as for the Centralization Alternative minimum option.

Option A, Suboption Y

- dry storage of all defense production fuel in a large vault facility
- transport of other Hanford Site fuel to INEL
- defense production fuel stabilized prior to storage
- use of existing facilities (FFTF and T Plant) and a stabilization facility to load shipping casks
- leakers, if any, unloaded in a special module at a stabilization facility.

For N Reactor and single-pass reactor fuel, this option is identical to the Decentralization Alternative; for other Hanford Site fuel, this option is nearly identical to the Centralization Alternative minimum option.

Option A, Suboption Z

- dry storage of all fuel in casks in a large facility
- defense production fuel stabilized prior to storage
- dry storage casks loaded at existing facilities (FFTF and T Plant)
- use of existing facilities (FFTF and T Plant) and a stabilization facility to load shipping casks
- leakers unloaded in a special module at a stabilization facility.

For N Reactor and single-pass reactor fuel, this option is identical to the Decentralization Alternative; for other Hanford Site fuel, this option is nearly identical to the Centralization Alternative minimum option.

Option B1

All fuel from offsite would be stored dry in casks in a large facility, although a very small amount might require wet storage for an interim period prior to dry storage. SNF received from other DOE locations would arrive stabilized and canned as necessary for storage. SNF received from universities and SNF of U.S. origin from foreign research locations would require canning prior to storage. The required receiving and canning would be done in a new facility because of the extended period over which the fuel would be received. A small amount of fuel would arrive after only limited time since reactor discharge, which would require temporary water storage until it aged sufficiently to be dry stored. That water storage would be included in the receiving and canning facility. Technology development would be conducted in a separate, nearby facility.

Option B2

The activities for this option would be the same as those for Option B1, except that additional storage would be required for Naval fuel.

Option C

Hanford fuel would be stabilized as necessary, loaded, and shipped offsite.

3.1.4.3 Existing Facilities. Upgrades, replacements, and additions to the existing facilities would occur as required under the Decentralization Alternative.

3.1.4.4 New Facilities. Research and development and pilot programs for characterization, stabilization, and other needs to support future decisions on the ultimate disposition of SNF would also occur. Refer to Table 3-6 for the potential facility requirements under the three storage and three stabilization options. A description of these options is given in Section 3.1.2.1, Anticipated Activities under the Decentralization Alternative. Options X, Y, and Z with their respective stabilization suboptions are the same as those for the Regionalization and Decentralization Alternatives (see Table 3-4). What is different is the specific assortment of fuel to be managed in each of the alternatives. The stabilization facilities required under the Regionalization Alternative are the same as those listed in Table 3-5.

Table 3-6. Description of required facilities under Regionalization Alternatives.

Alternatives	New Facility	Description	Capacity
Regionalization A/ Suboption X RAX	Water basin	Building: 109.7 m long x 42.7 m wide x 12.2 m high pre-cast concrete Land use: <8094 m^2 (<2 acres) Water storage pool: rectangular, 520 m^2, cast-in-place concrete Canisters: double barreled, each 0.23 m diameter x 0.74 m high Construction: 3-year deviation, operation starting in 2001	~2103 MTU in 8000 canisters
Regionalization A/ Suboption Y RAY	Shear/leach/calcine stabilization process	See Table 3-5	
Regionalization A/ Suboption RAY	Large modular dry storage vault	Building: 94.5 m long x 88.4 m wide x 18.3 m high cast-in-place concrete, pre-cast concrete superstructure Land Use: ~8094 m^2 (~2 acres) Canisters: 0.58 m diameter x 4.11 m high Construction: 3-year duration, operation to start in 2001	~2103 MTU in 1200 canisters
Regionalization A\ Suboption RAZ	Shear/leach/calcine stabilization process	See Table 3-5	
Regionalization A/ Suboption RAZ	Concrete storage module holding NUHOMs[a] casks	Building: 3.0 m wide x 5.5 m long x 4.6 m high Land Use: 16,187 m^2 (4 acres) Casks: 1.7 m diameter x 4.9 m long Construction: 3 year duration, operation to begin in 2001	2013 MTU in 230 prefabricated dry storage module casks

Table 3-6. (contd)

Alternatives	New Facility	Description	Capacity
Note: Facilities required for Alternatives RBl and RB2 are in addition to those required for Decentralization			
Regionalization B1, RB1	Incremental cask storage	Building: 121.9 m x 365.8 m Similar to but larger than that for Decentralization Option X	330 MTHM
	Receiving and canning facility	Building: 53.3 long x 53.3 m wide x 16.8 m high 3 foot thick cast-in-place concrete	188 shipping casks, 50 storage casks
	Technology development facility	Building: 53.3 m long x 30.5 m wide x 16.8 m high pre-cast concrete	
		Land use for all three RBl facilities: 40,469 m^2 (10 acres)	
		Construction: Receiving/canning and tech. dev. 1998-2001; for 90% of storage facility 2000-2010; for remaining 10% storage 2010-2035; operating period: 2000 through 2035	
Regionalization B2, RB2	Prefabricated by storage cask facility	Building: 914.4 m x 121.9 m; similar to but larger than Option X for Decentralization	400 MTHM (for total, with Decentralization, of 2500 MTHM)
	Receiving and canning facility	Sames as for RB1	188 shipping casks 50 storage casks
	Technology development facility	Same as for RB1	
	Land use for all three RB2 facilities: 101,172 m^2 (25 acres)		

a. NUHOMs casks [Nutech Horizontal Modular Storage (from Pacific Nuclear)]

3.1.5 Centralization Alternative

Under the Centralization Alternative for SNF storage, all current and future SNF from DOE and the Naval Nuclear Propulsion Program would be sent to one DOE site or other location. The activities at each site would depend on whether the SNF was being received or shipped offsite. Sites not selected would close down their storage facilities once the fuel had been removed. The following information summarizes the expected impact at Hanford and provides insight into the characteristics of the SNF and facilities that would be involved in shipping these fuels to Hanford.

3.1.5.1 Description of Spent Nuclear Fuel Inventory The SNF inventory that would exist at Hanford under this alternative would include that which is presently at Hanford (see Table 3-1), as well as any new fuel shipped to Hanford. If the minimum option occurs under the Centralization Alternative, then all of this spent fuel would be shipped offsite and there would no longer be a spent fuel inventory at Hanford, barring any required for research. If the maximum option occurs, the spent fuel at all of the other sites across the United States would eventually be transported to Hanford.

The locations from which spent fuel would be sent, in addition to SRS and INEL, include Argonne National Laboratories East and West, Babcock and Wilcox, Brookhaven National Laboratory, General Atomics, Los Alamos National Laboratory, Oak Ridge National Laboratory, Sandia National Laboratories, West Valley, and Fort St. Vrain. Naval spent nuclear fuel from shipyards and prototypes would be sent first to the equivalent of the Expended Core Facility, which would be relocated to Hanford. There the fuel would be examined by the Naval Nuclear Propulsion Program prior to being turned over to DOE for storage at Hanford. Foreign fuel that may be returned to the United States following irradiation or testing offsite would also be included in this inventory under the Centralization Alternative. Summaries of the spent fuel at each site are shown in Volume I, Attachments B, C, and D and Volume III of DOE (1993a). Additional information is in DOE (1992a) (Fort St. Vrain and Peach Bottom high-temperature gas-cooled reactor spent graphite fuel).

3.1.5.2 Anticipated Activities. If Hanford is chosen as the site for storing the entire spent fuel inventory, the upgrades, increases, and replacements of storage capacity would occur as required for the existing spent fuel as well as to accommodate the increased spent fuel inventory. If the Centralization Alternative is chosen and Hanford is not selected, the activities would include stabilization to ensure safe storage and transportation offsite.

All fuel received from offsite would be stored dry in casks in a large facility, although some may require wet storage for an interim period prior to dry storage. SNF received from other DOE sites will arrive stabilized and canned as necessary for storage. SNF received from universities and from foreign locations would require containerization prior to storage. Naval SNF would arrive uncontainerized, but would not require containerization. The required receiving and containerizing would be done in a new facility because of the large throughput involved and the extended period (40 years instead of 4) during which the fuel would be received. Some university and foreign fuel would require temporary wet storage. That water storage is included in the receiving and canning facility. Technology development would be conducted in a separate, nearby facility.

3.1.5.3 Description of New Facilities. The new facilities required for the alternative in which all U.S. DOE SNF would be stored at the Hanford Site are of the same type as, but larger than, those required for Regionalization Alternative Option B2:

- The Prefabricated Dry Storage Cask Facility for offsite SNF would be approximately 120 meters x 1200 meters.
- The Receiving and Canning Facility would be approximately 110 meters x 50 meters x 20 meters high.
- The Technology Development Facility would be approximately 50 meters x 40 meters x 20 meters high.
- The land required for these three facilities together would be approximately 14 hectares (35 acres).

3.2 Comparison of Alternatives

A summary of environmental impacts among the various alternatives is provided in Table 3-7. The alternatives are briefly described below to aid in interpreting the material presented.

The No Action Alternative identifies the minimum actions deemed necessary for continued safe and secure storage of SNF at the Hanford Site. Upgrade of the existing facilities would not occur other than as required to ensure safety and security.

The Decentralization Alternative includes additional facility upgrades over those considered in the No Action Alternative, specifically, new wet storage (for defense production fuel only) or dry storage facilities, fuel processing via shear/leach/calcination or shear/leach/solvent extraction, with research and development activities to support such processing.

The 1992/93 Planning Basis Alternative differs from the Decentralization Alternative only in that TRIGA fuel currently stored at the Hanford Site would be shipped offsite. The storage and stabilization options identified for the Decentralization Alternative are also assumed for the 1992/1993 Planning Basis Alternative.

The Regionalization Alternative as it applies to the Hanford Site consists of the following options:

- Option A (fuel type) - Defense production SNF would remain at Hanford; other types of fuel would be sent to INEL.
- Option B1 (geographic) - All SNF west of the Mississippi River, except Naval SNF would be sent to Hanford.
- Option B2 (geographic) - All SNF west of the Mississippi River and Naval SNF would be sent to Hanford.
- Option C (geographic) - All Hanford SNF would be sent to INEL or NTS.

Table 3-7. Summarized comparisons of the alternatives[a].

Resource or Consequence	Alternatives: No Action	Decentralization	1992/1993 Planning Basis	Regionalization A	Regionalization B1	Regionalization B2	Centralization at Hanford	Regionalization C and Centralization Elsewhere
Traffic and transportation	No change in onsite traffic patterns. Total population dose would be less than one person-rem and no fatal cancers would be projected.	From 1 to 6 percent increase in onsite traffic depending on suboption selected. Total population dose would be less than 2 person-rem and no fatal cancers would be projected.		From 1 to 5% increase in onsite traffic depending on suboption selected. Total population dose less than 1 person-rem and no fatal cancers would be projected.	Essentially same as Decentralization Alternative	Essentially same as Decentralization Alternative	Essentially same as Decentralization Alternative.	Onsite traffic not significantly different from No Action Alternative. Essentially no change Total population dose would be about 4 person-rem and no fatal cancers would be projected.
Health & Safety (fatal cancers over 40 years of normal operations)								
Occupational	None (0.4)	None (0.04-0.1)	None (0.04-0.1)	None (0.04-0.1)	None (0.3-0.4)	None (0.3-0.4)	None (0.4)	None (0.08)
Public (max)	None (5.2×10^{-4})	None (2.5×10^{-3})	None (2.5×10^{-3})	None (2.5×10^{-3})	None (2.5×10^{3})	None (2.5×10^{-3})	None (2.5×10^{-3})	None (2.5×10^{-3})
Utilities and energy (megawatt-hrs/yr) electrical[b]	12,000	100-127,000	100-127,000	100-127,000	100-127,000	100-127,000	100-127,000	0-20,000
Materials and waste management								
LLW, m^3/y	95	41-420	41-420	61-420	43-430	43-430	110-490	140-420
TRU waste, m^3/y	0	0-50	0-50	0-50	0-50	0-50	0-50	0-50
HLW, m^3/y	0	0-57	0-57	0-57	0-57	0-57	0-57	0-57
Mixed waste, m^3/y	1	0.23-2.10	0.23-2.0	0.23-2.0	0.26-2.0	0.26-2.0	0.51-2.3	1.0-2.0
Hazardous Waste, m^3/y	2.3	1.1-2.8	1.1-2.8	1.1-2.8	1.2-2.9	1.2-2.9	2.3-3.9	1.4-2.8

a. Hyphenated numbers indicate range of values depending on processing options selected

b. Minimum value represents requirements during the period after all fuel has been placed into dry storage or has been shipped offsite. Maximum value represents requirements during the interim period (less than 4 years) while SNF is being processed and prepared for storage or shipment offsite, assuming concurrent operation of the process facility and the existing facilities where SNF is currently stored (as in the No Action Alternative).

c. Spent filters and ion exchange resins are the only sources of TRU waste. Filters and resins are charged before they become TRU waste.

Table 3-7. (contd)

Resource or Consequence	Alternatives: No Action	Decentralization	1992/1993 Planning Basis	Regionalization A	Regionalization B1	Regionalization B2	Centralization at Hanford	Regionalization C and Centralization Elsewhere
Postulated Accidents								
Facilities								
Point estimate of fatal cancer risk - worst consequences accident - public	<3.7 x 10^{-3}	4.9 x 10^{-4}	4.9 x 10^{-4}	4.9 x 10^{-4}	5.7 x 10^{-4}	5.7 x 10^{-4}	6.5 x 10^{-4}	4.1 x 10^{-4}
Workers	<1.4 x 10^{-7}	5.6 x 10^{-7}	5.6 x 10^{-7}	5.6 x 10^{-7}	6.6 x 10^{-7}	6.6 x 10^{-7}	7.5 x 10^{-7}	4.7 x 10^{-7}
Transportation								
Numbers of fatal cancers	None (5.5 x 10^{-2})	1(0.7)	1(0.7)	None (6.8 x 10^{-2})	1(0.7)	1(0.7)	1(0.7)	1(0.7)
Land use (area converted for SNF stabilization, packaging and/or storage)	No change	4 to 7 ha (11-18 acres)	4 to 7 ha (11-18 acres)	4 to 7 ha (11-18 acres)	15-17 ha (36-43 acres)	25-28 ha (61-68 acres)	35-38 ha (86-93 acres)	2 to 5 ha (6-12 acres)
Socioeconomics (worker-years over 10 years)	No change	798-6374	798-6374	618-4684	1716-7592	2088-8039	2857-9019	3905-5846
Cultural Resources	No change	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected
Aesthetic and scenic	No change	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected
Geologic resources	No change	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected
Air quality and related consequences (fatal cancers over 40 years normal operations)	No change	None	None	None	None	None	None	None
Water quality and related consequences	Maximum radiological and non-radiological carcinogenic risks less than one chance per billion	Maximum radiological and nonradiological carcinogenic risks less than 50 chances per billion						No effects expected
Ecological resources (Habitat area destroyed)	No change	4 to 7 ha (11-18 acres)	4 to 7 ha (11-19 acres)	4 to 7 ha (11-18 acres)	15 to 17 ha (36-43 acres)	25-28 ha (61-68 acres)	35-38 ha (86-93 acres)	2 to 7 ha (6-12 acres)
Noise	No change	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected	No effects expected

Two options exist at the Hanford Site for the Centralization Alternative: 1) the minimum option, in which all SNF on the Hanford Site would be shipped offsite, and 2) the maximum option, in which all SNF within the DOE complex would be shipped to the Hanford Site for management and storage. In the latter case, dry storage of all fuel sent to the Hanford Site from offsite would be assumed. A facility equivalent to the Decentralization suboptions would be assumed for stabilization of defense production fuel prior to storage; fuel received from offsite would have been stabilized for dry storage prior to receipt.

4. AFFECTED ENVIRONMENT

4.1 Overview

The Hanford Site is characterized by a shrub-steppe climate with large sagebrush dominating the vegetative plant community. Jack rabbits, mice, badgers, deer, elk, hawks, owls, and many other animals inhabit the Hanford Site. The nearby Columbia River supports one of the last remaining spawning areas for Chinook salmon and hosts a variety of other aquatic life. The climate is dry with hot summers and usually mild winters. Severe weather is rare. With construction of dams along the Columbia River, flooding is nearly nonexistent.

The Hanford Site was a major contributor to national defense during World War II and the Cold War era. The site was selected because it was sparsely settled and the Columbia River provided an abundant supply of cold, clean water to cool the reactors. As a result of wastes generated by these national defense activities, there are presently more than 1500 waste management units and four major groundwater contamination plumes. These have been grouped into 78 operable units: 22 in the 100 Area (reactor area), 43 in the 200 Area (chemical processing and refining areas), 5 in the 300 Area (research and development area), and 4 in the 1100 Area (storage area). An additional four units are found in the 600 Area (the rest of the Hanford Site). Each of these operable units is following a schedule for clean-up established by the Hanford Federal Facility Agreement and Consent Order (Tri-Party Agreement), which involves the U.S. Department of Energy (DOE), the Washington Department of Ecology, and the EPA.

4.2 Land Use

A brief description of the existing land use on the Hanford Site and adjacent lands and a brief discussion devoted to the existing land use on the proposed project site area follow.

4.2.1 Land Use at the Hanford Site

The Hanford Site is used primarily by DOE. Public access is limited to travel on the two access roads as far as the Wye Barricade, on Highway 240, and on the Columbia River (see Figure 4-1). The site encompasses 1450 square kilometers (560 square miles), of which most is

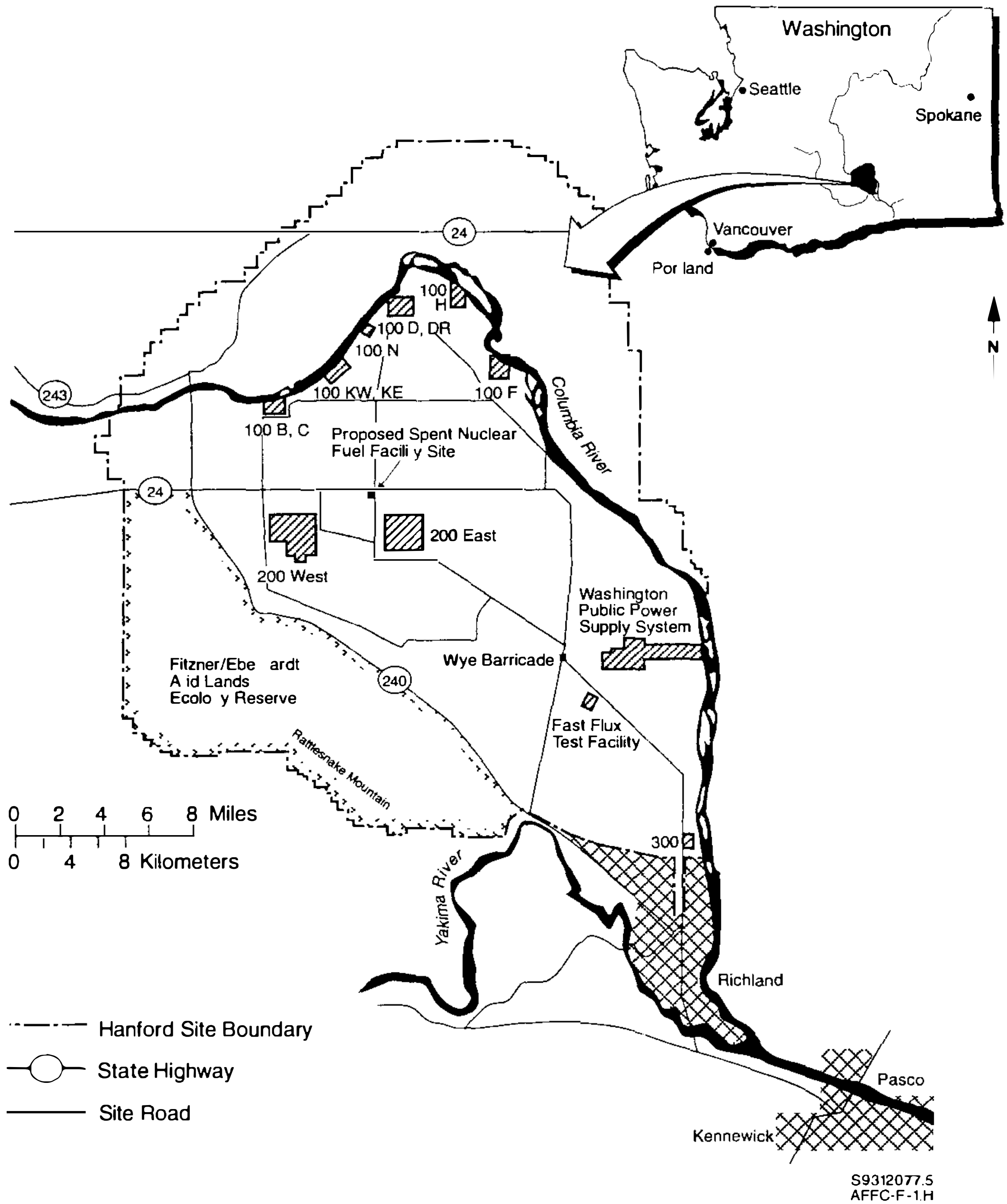

Figure 4-1. Hanford Site showing proposed spent nuclear fuel facility location.

open vacant land with widely scattered facilities, old reactors, and processing plants (Figure 4-1). In the past, DOE has stated that it intends to maintain active institutional control of the Hanford Site in perpetuity (DOE 1989). In the future, DOE could release or declare excess portions of the Hanford Site not required for DOE activities. Alternatively, Congress could act to change the management or ownership of the Hanford Site. The DOE operational areas are described below:

- The 100 Area [11 square kilometers (4.2 square miles)], which borders the right bank (south shore) of the Columbia River, is the site of eight retired plutonium production reactors and N Reactor, which is in shutdown deactivation status.
- The 200-West and 200-East Areas [16 square kilometers (6.2 square miles)] are located on a plateau about 8 and 11 kilometers (5 and 7 miles), respectively, from the Columbia River. These areas have been dedicated for some time to fuel reprocessing and waste processing management and disposal activities. The proposed project would be located between these areas.
- The 300 Area [1.5 square kilometers (0.6 square miles)], located just north of the city of Richland, is the site of nuclear research and development.
- The 400 Area [0.6 square kilometers (0.25 square miles)] is about 8 kilometers (5 miles) north of the 300 Area and is the site of the Fast Flux Test Facility (FFTF) used in the testing of breeder reactor systems. Also included in this area is the Fuels and Material Examination Facility.
- The 600 Area comprises the remainder of the Hanford Site and includes the Arid Land Ecology Reserve (ALE) [310 square kilometers (120 square miles)], which has been set aside for ecological studies, and the following facilities and sites:
    - a commercial low-level radioactive waste disposal site [4 square kilometers (1.7 square miles)], part of which is leased by the State of Washington.
    - Washington Public Power Supply System nuclear power plants [4.4 square kilometers (1.7 square miles)].
    - a 2.6-square kilometer (1 square mile) parcel of land transferred to Washington State as a potential site for the disposal of nonradioactive hazardous wastes.
    - a wildlife refuge of about 130 square kilometers (50 square miles) under revocable use permit to the U.S. Fish and Wildlife Service.
    - an area of about 6 square kilometers (2.3 square miles) has been provided to site a National Science Foundation Laser Gravitational-Wave Interferometer Observatory west of the 400 Area. When completed, this facility will occupy about 0.6 square kilometers (0.2 square miles).

- a recreational game management area of about 225 square kilometers (87 square miles) under revocable use permit to the Washington State Department of Game.
- support facilities for the controlled access areas.

In addition, an area comprising 310 square kilometers (120 square miles) has been designated for use as the ALE by the U.S. Fish and Wildlife Service for a wildlife refuge and by the Washington State Department of Wildlife for a game management area (DOE 1986a). The entire Hanford Site has been designated a National Environmental Research Park.

The Columbia River adjacent to the Hanford Site is a major site for public use by boaters, water skiers, fishermen, and hunters of upland game birds and migratory waterfowl. Some land access along the shore and on certain islands is available for public use.

4.2.2 Land Use in the Vicinity of the Hanford Site

Land use adjacent to the Hanford Site to the southeast and generally along the Columbia River includes residential, commercial, and industrial development. The cities of Richland, Kennewick, and Pasco are located along the Columbia River and are the closest major urban land uses adjacent to the Hanford Site. These cities (known as the Tri-Cities) together support a population of approximately 96,000.

Irrigated orchards and produce crops, dry-land farming, and grazing are also important land uses adjacent to the Hanford Site. In 1985 wheat represented the largest single crop in terms of area planted in Benton and Franklin counties with 190 square kilometers (73 square miles). Corn, alfalfa, hay, barley, and grapes are other major crops in Benton and Franklin counties. In 1986 the Columbia Basin Project, a major irrigation project to the north of the Tri-Cities, produced gross crop returns of $343 million, representing 19 percent of all crops grown in Washington State. In 1986 the average gross crop value per irrigated acre was $664.00. The largest percentage of irrigated acres produced alfalfa hay, 29.4 percent of irrigated acres; wheat, 15.0 percent; and corn (feed grain), 9.4 percent. Other significant crops are potatoes, apples, dried beans, asparagus, and pea seed.

4.2.3 Potential Project Land Use

The potential project site (Centralization Alternative) is located between the 200-West and 200-East Areas. The land is currently vacant. The proposed project would consist of constructing an SNF facility on the site. This potential project would involve typical land uses that occur during construction phases and a more industrial/commercial land use after reaching the operational stage.

4.2.4 Native American Treaty Rights

In prehistoric and early historic times, the Hanford Reach of the Columbia River was populated by Native Americans of various tribal affiliations. The Wanapum and the Chamnapum bands of the Yakama[a] tribe lived along the Columbia River from south of Richland upstream to Vantage (Relander 1986; Spier 1936). Some of their descendants still live nearby at Priest Rapids Dam (the Wanapum Tribe); others have been incorporated into the Yakama and Umatilla reservations. Palus people, who lived on the lower Snake River, joined the Wanapum and Chamnapum to fish the Hanford Reach of the Columbia River, and some inhabited the river's east bank (Relander 1986; Trafzer and Scheuerman 1986). Walla Walla and Umatilla people also made periodic visits to fish in the area. These people retain traditional secular and religious ties to the region, and many, young and old alike, have knowledge of the ceremonies and lifeways of their aboriginal culture. The Washane, or Seven Drums religion, which has ancient roots and had its start on what is now the Hanford Site, is still practiced by many people on the Yakama, Umatilla, Warm Springs, and Nez Perce reservations. Native plant and animal foods, some of which can be found on the Hanford Site, are used in the ceremonies performed by sect members.

Native American Lands designated on the Hanford Site fall under the protective rights of the Treaty of 1855 and the National Historic Preservation Act; these will be addressed further in the Cultural Resources Section. Under the Treaties of 1855, lands now occupied by the Hanford Site and other southeastern Washington lands were ceded to the United States by the confederated tribes and bands of the Yakama Indian Nation, the Confederated Tribes of the Umatilla Indian Reservation, and the Nez Perce Tribe. Under these treaties, the Native American tribes obtained the right to perform certain activities on those lands, including the

a. The spelling Yakama rather than Yakima has been adopted by the Yakama Nation.

rights to hunt, to fish at all usual and accustomed places and to erect temporary buildings for curing fish, to gather roots and berries, and to pasture horses and cattle on open unclaimed lands. The Wanapum Tribe, although members never signed a treaty, claims similar rights on ceded lands along the Columbia River.

Tribal members have expressed an interest in renewing their use of these resources in accordance with the Treaty of 1855, and the DOE is assisting them in this effort. Certain landmarks, especially Rattlesnake Mountain, Gable Mountain, Gable Butte, Goose Egg Hill, and various sites along the Columbia River, are sacred to them. The many cemeteries found along the river are also considered to be sacred.

4.3 Socioeconomics

Activity on the Hanford Site plays a dominant role in the socioeconomics of the Tri-Cities (Richland, Pasco, and Kennewick) and other parts of Benton and Franklin counties. The Tri-Cities serves as a market center for a much broader area of eastern Washington, including Adams, Columbia, Grant, Walla Walla, and Yakima counties. The Tri-Cities also serves parts of northeastern Oregon, including Morrow, Umatilla, and Wallowa counties. Socioeconomic impacts of changes at Hanford are mostly confined to the immediate Tri-Cities community and Benton and Franklin counties (Yakima County to a lesser extent). However, because of the significance of the wider agricultural region and surrounding communities in the Tri-Cities' economic base, this section briefly discusses the wider region as well. Detailed analyses of the socioeconomics are found in Scott et al. (1987) and Watson et al. (1984). Additionally, the impact of the proposed SNF facility might be altered by changes in socioeconomic resources in the surrounding counties of Adams, Columbia, Grant, Walla Walla, and Yakima in Washington state; and Morrow, Umatilla, and Wallowa counties in Oregon (these and Benton and Franklin counties comprise the designated region of influence; see Figure 4-2). This section describes the population, economic activity, housing, and public services and public finance of each county within the region of influence and the Tri-Cities. Because Benton and Franklin counties are expected to be most impacted from changes in Hanford Site activities, the information presented in this section concentrates on those counties, with less attention paid to the other areas within the defined region of influence.

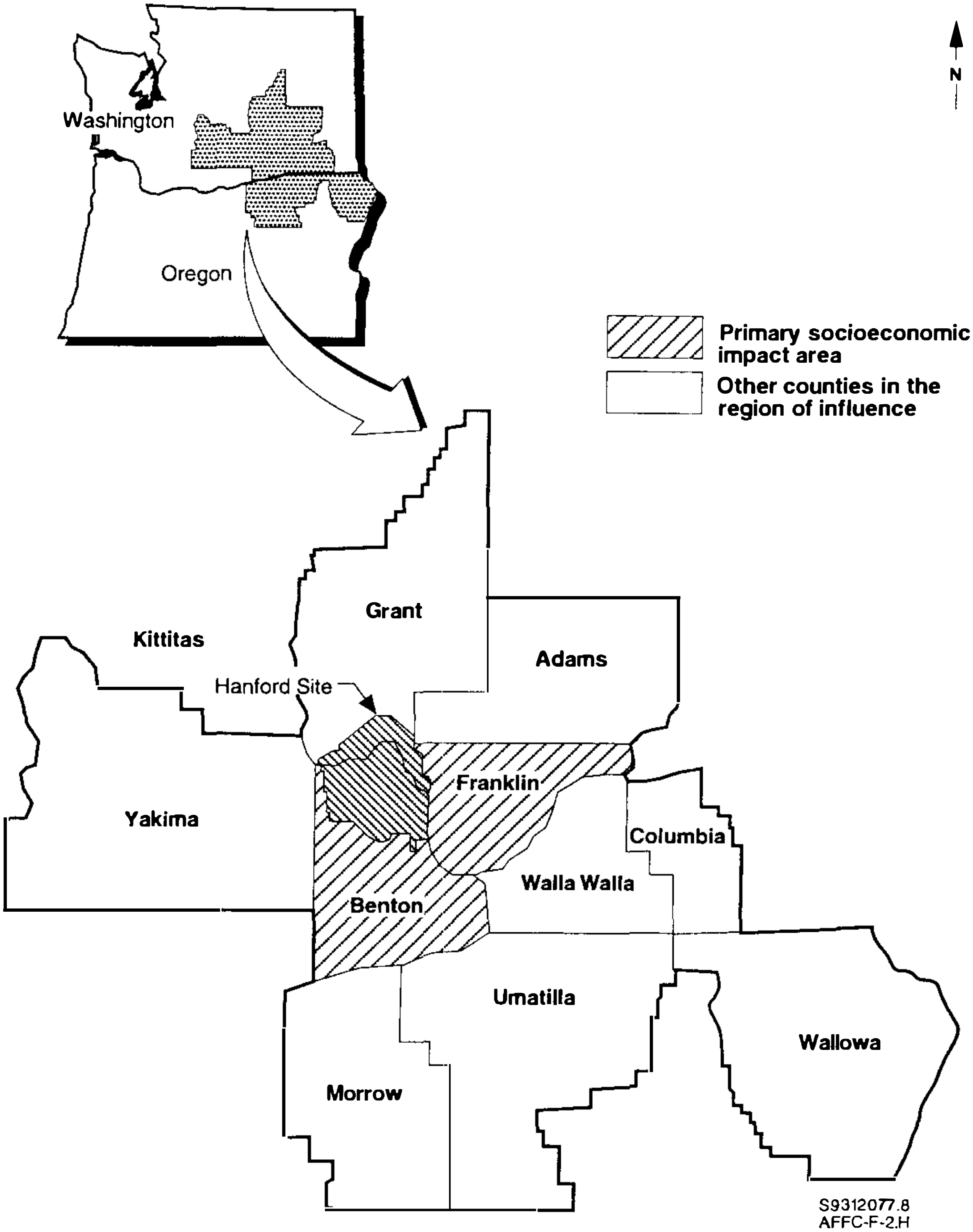

Figure 4-2. Areas of Washington and Oregon where socioeconomic resources may be affected by the proposed spent nuclear fuel facility (designated region of influence).

Table 4.3-1 summarizes the regional (Benton and Franklin counties) projections for employment, labor force, population, and Hanford Site employment by year for the years 1995-2004. Population projections were provided by the Washington State Office of Financial Management (1992a); employment projections were based on projections from the U.S. Department of Commerce (1992); labor force projections were based on an historical average unemployment rate of 8.8%; and Hanford Site employment projections were provided by DOE. It is anticipated at the time of this writing that a down-turn in Hanford Site employment will occur. The extent of the down-turn is unknown.

4.3.1 Demographics

This subsection briefly summarizes pertinent demographic information for each of the counties within the region of influence. Data for Washington were provided by the U.S. Department of Commerce (1992) and the Washington State Office of Financial Management (1992a,b). Data for Oregon were provided by the U.S. Department of Commerce (1992) and the Center for Population Research and Census (1993). Table 4.3-2 summarizes the population figures from 1960 to 1992 for each of the affected counties.

During the period from 1980 to 1990, growth in the affected Washington counties has been less than that of the state, with growth in the counties ranging from -0.07 percent (Columbia County) to 1.22 percent (Grant County) per year. During this same period, annual growth for the state of Washington averaged 1.66 percent. Washington counties within the region of influence also tended to have a younger population, with median ages ranging from 28.7 years to 39.0 years, as compared to the state median age of 33.1 years. These counties also tended to have a larger average household size than the state average, ranging from 2.44 to 3.03 persons, while the state average household size was listed at 2.53 persons.

Table 4.3-3 summarizes population projections through 2005 for each of the counties within the region of influence. All of the Washington counties are expected to experience continued growth, although most have projected growth rates less than that of the state. Washington is projected to have an increase in population of 21.8 percent by 2005 (from 4,866,692 in 1990 to 5,925,888 in 2005) for an annual average increase of 1.45 percent. Growth in the Oregon

Table 4.3-1. Regional economic and demographic indicators.

Year:	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Regional Employment	81,000	81,780	82,570	83,360	84,170	84,900	85,320	85,740	86,170	86,590
Regional Labor Force	88,820	89,670	90,540	91,410	92,290	93,090	93,550	94,020	94,480	94,950
Regional Population	162,660	164,810	166,980	169,180	171,410	173,380	175,730	178,100	180,510	182,950
Site Employment	18,700	16,200	14,700	14,700	14,700	14,700	14,700	14,700	14,700	14,700

Table 4.3-2. Population figures by county in the designated region of influence.

County	1960	1970	1980	1990	1992	1990 Median Age	1990 Average Household Size
Adams	9,929	12,014	13,267	13,603	14,100	30.7	2.94
Benton	62,070	67,540	109,444	112,560	118,500	32.1	2.65
Columbia	4,569	4,439	4,057	4,024	4,000	39.0	2.44
Franklin	23,342	25,816	35,025	37,473	39,200	28.7	3.03
Grant	46,477	41,881	48,522	54,758	58,200	31.9	2.74
Walla Walla	42,195	42,176	47,435	48,439	50,500	33.5	2.50
Yakima	145,112	145,212	172,508	188,823	193,900	31.5	2.80
Morrow	4,871	4,465	7,519	7,625	8,092[a]	—[b]	—
Umatilla	44,352	44,923	58,861	59,249	60,150[a]	—	—
Wallowa	7,102	6,247	7,273	6,911	7,135[a]	—	—

a. 1991 estimate.
b. Dash indicates the information was not available.

Table 4.3-3. Population projections by county in the designated region of influence.

County	1995 Forecast	1990 - 1995 % Change	2000 Forecast	1995 - 2000 % Change	2005 Forecast	2000 - 2005 % Change
Adams	13,867	1.94	14,163	2.14	14,424	1.84
Benton	121,328	7.79	128,752	6.12	136,892	6.32
Columbia	4,025	0.03	4,037	0.30	4,074	0.90
Franklin	41,336	10.31	44,630	7.97	48,213	8.03
Grant	58,026	5.97	60,518	4.30	62,983	4.07
Walla Walla	49,047	1.26	49,910	1.76	50,891	1.97
Yakima	199,578	5.70	207,870	4.15	216,245	4.03
Morrow	8,095	6.16	8,596	6.19	9,157	6.53
Umatilla	62,658	5.75	66,056	5.42	69,506	5.22
Wallowa	7,065	2.23	7,253	2.66	7,496	3.35

counties within the region of influence occurred rapidly during the 1970s; however, since 1980 population growth has tapered off. The Oregon counties within the region of influence are also expected to experience continued growth, although all have projected growth rates less than that of the state. Oregon is projected to have an increase in population of 25.5 percent (from 2,842,321 in 1990 to 3,566,189 in 2005) by 2005 for an annual average increase of 1.70 percent.

Within Benton and Franklin counties, the 1992 estimates distributed the Tri-Cities population as follows: Richland, 33,550; Kennewick, 44,490; and Pasco, 20,840. The combined populations of Benton City, Prosser, and West Richland totaled 10,460 in 1992. The unincorporated population of Benton County was 30,000. In Franklin County, incorporated areas other than Pasco had a total population of 2,540. The unincorporated population of Franklin County was 15,820.

4.3.2 Economics

This subsection summarizes pertinent economic activity within the region of interest and the Tri-Cities, including information on the general economy, employment, income, and impact of the Hanford Site. Historically, the primary industries within the region of influence have been related to agriculture; a multitude of crops encompassing many fruits, vegetables, and grains, are grown each year. Nearly all of the counties in the region of influence are home to food processing industries. Other primary industries within the region of influence include those relating to the wood industry: lumber, wood, and paper products. The data source for the Washington counties was the 1993 Washington State Yearbook (Office of the Secretary of State 1993), and the data source for the Oregon counties data was the 1991-92 Oregon Blue Book (Office of the Secretary of State 1991). Table 4.3-4 summarizes the primary industries, total employment for 1990, and total payroll for 1990 for the region of influence.

4.3.2.1 Employment in the Region of Interest. This subsection provides information on the employment and payroll breakdown by sector for each county within the region of influence. The source for the Washington counties was Washington State Employment Security Office (1992). The source for the Oregon counties was Department of Human Resources (1990). Tables 4.3-5 and and 4.3-6 provide information on average employment and payroll for 1990, broken down by

Table 4.3-4. County economic summary.

County	Primary Industries	1990 Total Employment	1990 Total Payroll ($ Million)
Adams	Food processing, agriculture	6,142	87.2
Benton	Food processing, chemicals, metal products, nuclear products	50,216	1,200.0
Columbia	Agriculture, food processing, wood products	1,559	22.3
Franklin	Food processing, publishing, agriculture, metal fabrication	17,958	284.6
Grant	Food processing, agriculture	20,851	346.0
Walla Walla	Food processing, agriculture, wood and paper products, manufacturing	20,546	366.5
Yakima	Agriculture, food processing, wood products, manufacturing	82,706	1,300.0
Morrow	Agriculture, food processing, utilities, lumber, livestock, recreation	2,791	53.5
Umatilla	Agriculture, food processing, wood products, tourism, manufacturing, recreation	21,448	366.0
Wallowa	Agriculture, livestock, lumber, recreation	2,216	37.9

industry, for each of the counties within the region of influence. For the Washington counties, the average employment includes only persons covered by the Employment Security Act and federal employment covered by Title 5, USC 85. For the Oregon counties, average employment includes only employees of businesses covered by the Employment Division Law.

4.3.2.2 Employment in the Tri-Cities. Three major sectors have been the principal driving forces of the economy in the Tri-Cities since the early 1970s: (1) the DOE and its contractors, which operate the Hanford Site; (2) Washington Public Power Supply System in its construction and operation of nuclear power plants; and (3) agriculture, including a substantial food-processing industry. With the exception of a minor amount of agricultural commodities sold to local area consumers, the goods and services produced by these sectors are exported from the Tri-Cities. In addition to direct employment and payrolls, these major sectors also support a sizable number of jobs in the local economy through their procurement of equipment, supplies, and business services.

Table 4.3-5. Employment by industry in the region of influence, 1990 figures.

Industry	Adams	Benton	Columbia	Franklin	Grant	Morrow	Umatilla	Walla Walla	Wallowa	Yakima
Agriculture, Forestry, Fisheries	1,660	4.487	105	4,265	4.496	558	1.366	1.890	54	20,342
Mining	0	3	0	89	0	0	0	0	0	641
Construction	0	2,809	27	628	0	33	592	0	86	2,427
Manufacturing	1036	12,310	563	1,599	2,761	884	4,654	3,993	509	9,671
Transportation and Public Utilities	236	884	58	1,212	657	153	899	593	85	2,824
Wholesale Trade	581	932	57	1,279	1,156	70	1,201	760	76	7,101
Retail Trade	720	7,865	120	2,669	3,109	195	3,845	3,639	360	12,537
Finance, Insurance, Real Estate	120	1,342	24	358	432	50	590	718	82	1,904
Services	564	11,741	144	2,768	2,512	142	3,416	4,207	204	14.491
Government	1,132	7,843	461	3,091	4,618	697	4,823	4,308	739	11,368
Not Elsewhere Classified	93	0	0	0	1,110	8	63	438	23	0

Table 4.3-6. Payroll by industry in the region of influence, 1990 figures ($ million).

Industry	Adams	Benton	Columbia	Franklin	Grant	Walla Walla	Yakima	Morrow	Umatilla	Wallowa
Agriculture, Forestry, Fisheries	14.7	39.1	1.5	39.1	47.9	18.4	173.4	9.0	18.7	0.7
Mining	0	0.1	0	2.3	0	0	0.6	0	0	0
Construction	0	79.3	1.0	12.7	0	0	47.7	0.5	11.9	2.1
Manufacturing	19.6	443.9	7.3	28.4	59.7	94.0	205.2	19.3	88.2	11.2
Transportation and Public Utilities	3.9	21.2	1.2	25.1	14.4	14.1	62.5	6.2	19.6	1.6
Wholesale Trade	10.7	19.2	1.1	26.3	21.4	15.6	118.4	1.5	22.2	1.2
Retail Trade	7.1	89.0	1.0	31.5	30.3	36.1	143.0	1.5	41.8	3.8
Finance, Insurance, Real Estate	2.0	22.0	0.4	6.2	7.6	13.2	39.0	1.0	10.6	1.0
Services	6.3	286.4	1.2	42.2	28.0	66.6	226.1	1.3	48.3	2.2
Government	21.2	225.8	7.7	70.8	107.0	100.0	258.0	12.8	103.6	13.7
Not Elsewhere Classified	1.6	0	0	0	29.7	8.6	0	0.2	1.0	0.3

1) *The DOE and its Contractors (Hanford).* Hanford continued to dominate the local employment picture with almost one-quarter of the total nonagricultural jobs in Benton and Franklin counties in 1992 (16,100 of 67,300). Hanford's payroll has a widespread impact on the Tri-Cities economy and state economy in addition to providing direct employment. These effects are further described in Subsection 4.3.

2) *Washington Public Power Supply System.* Although activity related to nuclear power construction ceased with the completion of the WNP-2 reactor in 1983, the Washington Public Power Supply System continues to be a major employer in the Tri-Cities area. Headquarters personnel based in Richland oversee the operation of one generating facility and perform a variety of functions related to two mothballed nuclear plants and one standby generating facility. In 1992, the Washington Public Power Supply System headquarters employment was more than 1700 workers. Washington Public Power Supply System activities generated a payroll of approximately $80.4 million in the Tri-Cities during the year.

3) *Agriculture.* In 1990 agricultural activities in Benton and Franklin counties were responsible for approximately 12,900 jobs, or 17 percent of the area's total employment. According to the U.S. Department of Commerce's Regional Economic Information System, about 2200 people were classified as farm proprietors in 1990. Farm proprietors' income from this same source was estimated at $121 million in the same year.

Crop and livestock production in the bicounty area generated about 7600 wage and salary jobs in 1990, as represented by the employees covered by unemployment insurance. The presence of seasonal farm workers would increase the total number of farm workers. Apart from the difficulty of obtaining reliable information on the number of seasonal workers, however, is the question of how much of these earnings are actually spent in the local area. For this analysis, the assumption is that the impact of seasonal workers on the local economy is sufficiently small to be safely ignored.

The area's farms and ranches generate a sizable number of jobs in supporting activities, such as agricultural services (for example, application of pesticides and fertilizers or irrigation system development) and sales of farm supplies and equipment. These activities, often called agribusiness, are estimated to employ 900 people. Although formally classified as a manufacturing activity, food processing is a natural extension of the farm sector. More than 20 food processors in Benton and Franklin counties produce such items as potato products, canned fruits and vegetables, wine, and animal feed.

In addition to those three major employment sectors, three other components are readily identified as contributors to the economic base of the Tri-Cities economy. The first component, categorized as other major employers, includes five employers: (1) Siemens Nuclear Power Corporation in north Richland, (2) Sandvik Special Metals in Kennewick, (3) Boise-Cascade in Wallula, (4) Burlington Northern Railroad in Pasco, and (5) Iowa Beef Processors in Wallula. The second component is tourism. The Tri-Cities area has increased its convention business substantially in recent years, in addition to business generated by travel for recreation. The final component in the economic base relates to the local purchasing power generated from retired former employees. Government transfer payments in the form of pension benefits constitute a significant proportion of total spendable income in the local economy.

Retirees. Although the Benton and Franklin counties have a relatively young population (approximately 56 percent under the age of 35), 15,093 people over the age of 65 resided in Benton and Franklin counties in 1990. The portion of the total population that is 65 years and older is currently increasing at about the same rate as that being experienced by Washington State (3.0 percent and 3.1 percent, respectively). This segment of the population supports the local economy on the basis of income received from government transfer payments and pensions, private pension benefits, and prior individual savings.

Although information on private pensions and savings is not available, data are available regarding the magnitude of government transfer payments. The U.S. Department of Commerce's Regional Economic Information System has estimated transfer payments by various programs at the county level. A summary of estimated major government pension benefits received by the residents of Benton and Franklin counties in 1990 is shown in Table 4.3-7. About two-thirds of the Social Security payments go to retired workers; the remainder are for disability and other payments. The historical importance of government activity in the Tri-Cities area is reflected in the relative magnitude of the government employee pension benefits as compared to total payments.

Table 4.3-7. Government retirement payments in Benton and Franklin counties in 1990 ($ million).

Source	Benton County	Franklin County	Total
Social Security (including survivors and disability)	101.5	31.1	132.6
Railroad retirement	2.7	3.6	6.3
Federal civilian retirement	10.5	2.8	13.3
Veterans pension and military retirement	14.7	3.1	17.8
State and local employee retirement	22.3	5.5	27.8
Total	151.7	46.1	197.8

4.3.2.3 Income Sources. Three measures of income are presented in Table 4.3-8: total personal income, per capita income, and median household income. Total personal income is comprised of all forms of income received by tbe populace, including wages, dividends, and other revenues. Per capita income is roughly equivalent to total personal income divided by the number of people residing in the area. Median household income is the point at which half of the households have an income greater than the median and half have less. The source for total personal income and per capita income was the U.S. Department of Commerce's Regional Economic Information System; while median income figures for Washington State were provided in Washington State Office of Financial Management (1992b), and by personal communication with the Bureau of Census Housing Division for Oregon.

In 1990 the total personal income for the Washington was $92.2 billion; of this, the counties within the region of influence comprised 8.0 percent. Per capita income for Washington State was $18,777; all Washington counties within the region of influence had per capita incomes less than that of the state. All Washington counties within the region of influence, with the exception of Benton, had median household incomes less than the state median of $32,725.

In 1990 the total personal income for Oregon was $49.2 billion; of this, the counties within the region of influence comprised 2.4 percent. Per capita income for Oregon State was $17,182; two of the three affected Oregon counties had per capita incomes greater than that of the state in 1990; however, only one of the three counties had a median household income greater than the state median of $27,250.

Table 4.3-8. Income measures by county, 1990 figures.

County	Total Personal Income ($ Million)	Per Capita Income ($)	Median Income ($)
Adams	231	16,897	25,750
Benton	1,960	17,332	33,800
Columbia	72	17,927	21,000
Franklin	553	14,734	26,300
Grant	854	15,511	23,625
Walla Walla	799	16,438	25,400
Yakima	2,920	15,374	24,525
Morrow	144	18,868	29,969
Umatilla	896	15,069	22,791
Wallowa	121	17,461	21,300

4.3.2.4 Hanford Employment. In 1991 Hanford employment accounted directly for 24 percent of total nonagricultural employment in Benton and Franklin counties and slightly more than 0.6 percent of all statewide nonagricultural jobs. In 1991 Hanford Site operations directly accounted for an estimated 42 percent of the payroll dollars earned in the area.

Previous studies have revealed that each Hanford job supports about 1.2 additional jobs in the local service sector of Benton and Franklin counties (about 2.2 total jobs) and about 1.5 additional jobs in the state's service sector (about 2.5 total jobs) (Scott et al. 1987). Similarly, each dollar of Hanford income supports about 2.1 dollars of total local incomes and about 2.4 dollars of total statewide incomes. Based on these multipliers, Hanford directly or indirectly accounts for more than 40 percent of all jobs in Benton and Franklin counties.

Based on employee residence records as of December 1993, 93 percent of the direct employment of Hanford is comprised of residents of Benton and Franklin counties. Approximately 81 percent of the employment is comprised of residents who reside in one of the Tri-Cities. More than 42 percent of the employment is comprised of Richland residents, 30 percent of Kennewick residents, and 9 percent of Pasco residents. West Richland, Benton City, Prosser, and other areas in Benton and Franklin counties account for 12 percent of total employment. Table 4.3-9 contains the estimated percent of Hanford employees residing in each of the counties within the region of influence. The information available did not include the

Table 4.3-9. Hanford employee residences by county.

County	Percent of Employees in Residence
Adams	0.18%
Benton	84.16%
Columbia	0.01%
Franklin	9.07%
Grant	0.25%
Walla Walla	0.21%
Yakima	5.08%
Morrow	0.01%
Umatilla	0.01%

residences of DOE employees nor those of ICF Kaiser Hanford Company or the Bechtel Hanford Company. It was assumed that the distribution of these employees would be similar to the distribution of the other Hanford contractors.

Hanford and contractors spent nearly $298 million, or 45.6 percent of total procurements of $653 million, initially through Washington firms in 1993. About 18 percent of Hanford orders were filled by Tri-Cities firms.

Hanford contractors paid a total of $10.9 million in state taxes on operations and purchases in fiscal year 1988 (the most recent year available). Estimates show that Hanford employees paid $27.0 million in state sales tax, use taxes, and other taxes and fees in fiscal year 1988. In addition, Hanford paid $0.9 million to local government in Benton, Franklin, and Yakima counties in local taxes and fees (Scott et al. 1989).

4.3.3 Emergency Services

This subsection contains information on the law enforcement, fire protection, and health services provided by each county within the region of influence. These figures are presented in Table 4.3-10, with more detailed information about the Tri-Cities area. Law enforcement figures were obtained from each county sheriff's office in December 1993. Data on fire protection and health care facilities were provided by the Office of the Secretary of State (1993).

Table 4.3-10. Emergency services within the region of influence.

County	Commissioned Officers - County Sheriff	Number of Fire Districts - Unincorporated	Number of Hospitals
Adams	16 + Sheriff	7	2
Benton	40	6	3
Columbia	10 + Sheriff	3	1
Franklin	18 + Sheriff	4	1
Grant	35 + Sheriff	12	1
Walla Walla	16 + Sheriff	8	2
Yakima	63	12	3
Morrow	70	NA	NA
Umatilla	12	NA	NA
Wallowa	5	NA	NA

Police protection in Benton and Franklin counties is provided by the Benton and Franklin County sheriff's departments, local municipal police departments, and the Washington State Patrol Division headquartered in Kennewick. Table 4.3-11 shows the number of commissioned officers and patrol cars in each department in June 1992.

Table 4.3-11. Police personnel in the Tri-Cities in 1992.

Area	Commissioned Officers	Patrol Cars
Kennewick Municipal	58	32
Pasco Municipal	39	11
Richland Municipal	44	35
West Richland Municipal	7	9
County Sheriff, Benton County	43	50
County Sheriff, Franklin County	23	23

Source: Personal communication with each department office, January 1993.

The Kennewick, Richland, and Pasco municipal departments maintain the largest staffs of commissioned officers with 53, 44, and 38, respectively.

The Hanford Fire Department, composed of 126 firefighters, is trained to dispose of hazardous waste and to fight chemical fires. During the 24-hour duty period, five firefighters cover the 1100 Area, seven protect the 300 Area, seven watch the 200-East and 200-West Areas, six are responsible for the 100 Areas, and six cover the 400 Area, which includes the WPPSS area. To perform their responsibilities, each station has access to a Hazardous Material Response Vehicle that is equipped with chemical fire extinguishing equipment, an attack truck that carries foam and Purple-K dry chemical, a mobile air truck that provides air for gasmasks, and a transport tanker that supplies water to six brush-fire trucks. The Hanford Fire Patrol owns five ambulances and maintains contact with local hospitals.

Table 4.3-12 indicates the number of fire-fighting personnel, both paid and unpaid, on the staffs of fire districts in the Tri-Cities area.

The Tri-Cities area is served by three hospitals: Kadlec Hospital, Kennewick General, and Our Lady of Lourdes. In addition, the Carondelet Psychiatric Care Center is located in Richland. Kadlec Hospital, located in Richland, has 136 beds and functions at 39.5 percent

Table 4.3-12. Fire protection in the Tri-Cities in 1992[a].

Station	Fire-Fighting Personnel	Volunteers	Total	Service Area
Kennewick	54	0	54	City of Kennewick
Pasco	30	0	30	City of Pasco
Richland	50	0	50	City of Richland
BCRFD[b] 1	6	120	126	Kennewick Area
BCRFD 2	1	31	32	Benton City
BCRFD 4	4	30	34	West Richland

a. Source: Personal communication with each department office, January 1993.
b. BCRFD = Benton County Rural Fire Department.

capacity. Their 5754 annual admissions represent more than 42 percent of the Tri-Cities market. Non-Medicare/Medicaid patients accounted for 86 percent, or 4982 of their annual admissions. An average stay of 3.8 days per admission was reported for 1991.

Kennewick General Hospital maintains a 45.5 percent occupancy rate of its 71 beds with 3619 annual admissions. Non-Medicare/Medicaid patients in 1991 represented 58 percent of its total admissions. An average stay of 3.5 days per admission was reported.

Our Lady of Lourdes Health Center, located in Pasco, reported an occupancy rate of 36.5 percent; however, a significant amount of outpatient care is performed there. The outpatient income serves as a primary source of income for the center. In 1990 Our Lady of Lourdes had 3328 admissions, of which 52 percent were non-Medicare/Medicaid patients. The institution reported an average admission stay of 5.33 days.

4.3.4 Infrastructure

4.3.4.1 Housing. This section provides information on the total number of housing units, the number of occupied housing units, and a breakdown of total housing units by type for each of the counties within the region of influence. Additionally, specific information on the housing market in the Tri-Cities is included. The data source for Washington counties was the Washington State Office of Financial Management (1992b). The data source for the Oregon counties was hy personal communication with the Population Research Center at Portland State University. The data source for the Tri-Cities was by personal communication with the Washington State Office of Financial Management. Table 4.3-13 summarizes housing information by county for 1990 for the region of influence.

In 1993 nearly 94 percent of all housing (of 40,344 total units) in the Tri-Cities was occupied. Single-unit housing, which represents nearly 58 percent of the total units, had a 97 percent occupancy rate throughout the Tri-Cities. Multiple-unit housing, defined as housing with two or more units, had an occupancy rate of nearly 94 percent. Pasco had the lowest occupancy rate, 92 percent, in all categories of housing; followed by Kennewick, 95 percent, and Richland, 96 percent. Mobile homes, which represent 9 percent of the housing unit types, had

Table 4.3-13. Housing by county in 1990.

County	Total	Occupied	Vacancy Rate	Single Family	Multiple Family	Mobile Homes
Adams	5,263	4,586	12.9%	3,324	643	1,296
Benton	44,877	42,227	5.9%	28,193	10,592	6,092
Columbia	2,046	1,582	22.7%	1,597	146	303
Franklin	13,664	12,196	10.7%	7,782	3,289	2,593
Grant	22,809	19,745	13.4%	13,692	2,661	6,456
Walla Walla	19,029	17,623	7.4%	13,071	3,837	2,121
Yakima	70,852	65,985	6.9%	49,356	11,174	10,322
Morrow	3,412	2,803	17.8%	1,828	366	1,192
Umatilla	24,333	22,020	9.5%	15,178	4,503	4,418
Wallowa	3,755	2,796	25.5%	2,935	235	554

the lowest occupancy rate, 90 percent. In 1989 mobile homes had the highest occupancy rate, 93 percent. Table 4.3-14 shows a detailed listing of total units and occupancy rate by type in the Tri-Cities.

4.3.4.2 Human Services. The Tri-Cities offer a broad range of social services. State human service offices in the Tri-Cities include the Job Services office of the Employment Security Department; Food Stamp offices; the Division of Developmental Disabilities; Financial and Medical Assistance; the Child Protective Service; emergency medical service; a senior companion program; and vocational rehabilitation.

Table 4.3-14. Total units and occupancy rates (1993 estimates)[a].

City	All Units	Rate	Single Units	Rate	Multiple Units	Rate	Mobile Homes	Rate
Richland	14,388	96	9,921	98	3,827	95	640	88
Pasco	7,846	92	3,679	96	2,982	91	1,016	86
Kennewick	18,110	95	9,824	97	5,944	96	1,942	97
Tri-Cities	40,344	94	23,424	97	12,753	94	3,598	90

a. Source: Personal communication, Office of Financial Management, State of Washington, Forecast Division.

The Tri-Cities are also served by a large number of private agencies and voluntary human services organizations. The United Way, an umbrella fund-raising organization, incorporates 25 participating agencies offering more than 50 programs (United Way 1992).

4.3.4.3 Government. This subsection presents the county government revenues by source (Table 4.3-15) and expenditures by function (Table 4.3-16) for each of the counties within the region of influence. The data were taken from U.S. Department of Commerce (1990, 1993). All county data, with the exception of Benton and Yakima counties, are from 1986-87. Benton and Yakima county data are from 1990-91. These years were the most recent ones available.

4.3.4.4 Public Education. This subsection provides information on the educational sectors of each of the counties. The source for school district information, secondary education, and enrollment data for the Washington counties was the Office of the Secretary of State (1993); student/teacher ratios were provided by personal communication with the school districts. Information on the Oregon counties was provided by personal communication with the individual counties. Table 4.3-17 summarizes information on the number of school districts, enrollment, and post-secondary institutions within the region of influence.

In the Tri-Cities area, Benton County primary and secondary education is served by six school districts with an enrollment of 24,876 students in 1992. The student/teacher ratio in the Finley School District is 20.2; in Kennewick, 24.0; in Kiona Benton-City, 25.0; in Prosser, 22.0 for elementary and 25.0 for secondary; and in Richland, 23.0. The Paterson School District had an enrollment of 54 students in 1992, therefore a student/teacher ratio was not sought. Currently, the Kennewick, Richland, and Kiona-Benton City school districts are operating at or near capacity; Kennewick is working to alleviate some of the overcrowded conditions by constructing one new middle school and two new elementary schools. In addition, plans are under way for the construction of a new high school, scheduled to open in 1997. Kiona-Benton City is in the process of building additions at elementary and middle schools. The county also has a post-secondary institution located in Richland, a branch campus of Washington State University, WSU Tri-Cities. Enrollment for spring 1992 was 981 students.

Franklin County primary and secondary education is served by four school districts with an enrollment of 8,756 students in 1992 and a student/teacher ratio of 7.0 in Kahlotus; 17.6 in

Table 4.3-15. Revenue sources by county FY 1986-87 ($ thousand).

County	Total	Total	Intergovernmental revenue: From federal government	Intergovernmental revenue: From state government	General revenue from own sources: Total	General revenue from own sources: Taxes	Utility, liquor store, and employee retirement revenue
Adams	6,690	6,690	736	2,844	3,047	2,304	-[a]
Benton[b]	24,079	24,079	43	7,879	14,064	10,762	-
Columbia	2,560	2,560	78	1,388	1,040	720	-
Franklin	6,279	6,279	361	109	5,604	4,859	-
Grant	17,525	17,525	670	7,661	8,932	6,195	-
Walla Walla	11,698	11,698	426	3,763	7,008	5,658	-
Yakima[b]	45,310	45,289	392	14,066	28,864	20,429	21
Morrow	5,901	5,901	104	1,045	4,724	3,338	-
Umatilla	9,594	9,594	204	4,971	4,414	3,087	-
Wallowa	6,215	6,215	60	2,180	3,881	905	-

a. Dash indicates that the information was not available.
b. FY 1990-91.

Table 4.3-16. Expenditures by county FY 1986-87 ($ thousand).

		General Expenditures													
				Major Functions											
County	Total	Total	Capital Outlay	Education	Welfare	Hospitals	Health	Highways	Police protection	Correction	Natural resources and parks and recreation	Sewage and sanitation	Interest on general debt	Utility, liquor store, and employee retirement expenditure
Adams	6431	6431	1007	13	-[a]	-	286	3591	475	297	138	184	22	-
Benton[b]	22027	22027	890	9	-	-	3626	3190	1956	4129	216	-	223	-
Columbia	2647	2647	255	-	-	-	230	1106	265	13	306	84	-	-
Franklin	8230	8230	608	-	-	-	461	2883	855	811	177	-	49	-
Grant	17589	17589	3314	-	-	-	1403	6617	1443	1180	704	412	22	-
Walla Walla	11879	11879	432	4	-	-	1068	4624	1257	610	766	143	-	-
Yakima[b]	45967	45937	10059	-	187	-	989	9761	4188	7382	2971	415	487	30
Morrow	6382	6382	411	216	349	1113	325	1860	270	98	237	-	-	-
Umatilla	10707	10707	188	1095	-	-	2562	2337	540	561	346	-	-	-
Wallowa	6139	6139	362	339	794	2070	143	1181	208	111	198	67	9	-

a. Dash indicates that the information was not available.
b. FY 1990-91.

Table 4.3-17. Educational services by county in 1992.

County	Number of School Districts	Enrollment (1992)	Post-Secondary Education Institutions
Adams	5	3,437	0
Benton	6	24,876	1
Columbia	2	750	0
Franklin	4	8,756	1
Grant	10	13,232	1
Walla Walla	7	8,324	3
Yakima	15	42,227	3
Morrow	1	2,008[a]	0
Umatilla	12	12,500[a]	1
Wallowa	3	1,408[a]	0

a. 1993 enrollment

North Franklin; and 18.1 in Pasco. The Star School District had an enrollment of 15 students in 1992; therefore, a student/teacher ratio was not sought. Currently, Pasco School District is operating at or near capacity; however, the district is in the process of remodeling an old high school. The county also has a post-secondary institution of learning in Pasco, Columbia Basin Community College. Enrollment for 1992 was 6424 students.

4.4 Cultural Resources

The Hanford Site is known to be rich in cultural resources. It contains numerous, well-preserved archaeological sites representing both the prehistoric and historical periods and is still thought of as a homeland by many Native American people. A total of 248 known sites are prehistoric, 202 are historic, and 14 sites contain both prehistoric and historic components. Management of Hanford's cultural resources follows the Hanford Cultural Resources Management Plan (Chatters 1989) and is conducted by the Hanford Cultural Resources Laboratory of Pacific Northwest Laboratory (PNL). The Plan contains contingency guidelines for handling the discovery of previously unknown cultural resources encountered during construction activities.

Cultural resources are defined as any prehistoric or historic district, site, building, structure, or object considered to be important to a culture, subculture, or community for scientific, traditional, religious or any other reason. These are usually divided into three major categories: prehistoric and historic archaeological resources, architectural resources, and traditional cultural resources. Significant cultural resources are those that are eligible or potentially eligible to the National Register of Historic Places (36 CFR 60.4).

Consultation is required to identify traditional cultural properties that are important to maintaining the cultural heritage of Native American Tribes. Under the Treaties of 1855, lands ultimately occupied by the Hanford Site were ceded to the United States by the confederated tribes and bands of the Yakama Indian Nation, and Confederated Tribes of the Umatilla Indian Reservation. Under the treaty, the Native American Tribes acquired the rights to perform certain activities on open unclaimed lands, including the rights to hunt, fish, gather foods and medicines, and pasture livestock on these lands. By the time the Hanford Site was established, little open unclaimed land remained. The Wanapum Band and the Joseph Band of the Nez Perce Tribes never signed a treaty but have cultural ties to these lands.

The methodology for identifying, evaluating, and mitigating impacts to cultural resources is defined by federal laws and regulations including the National Historic Preservation Act (NHPA), the Archaeological Resource Protection Act (ARPA), the Native American Graves Protection and Repatriation Act (NAGPRA) and the American Native American Religious Freedom Act (AIRFA). A project affects a significant resource when it alters the property's characteristics, including relevant features of its environment or use, that qualify it as significant according to the National Register criteria. These effects may include those listed in 36 CFR 800.9. Impacts to traditional Native American properties can be determined only through consultation with the affected Native American groups.

4.4.1 Prehistoric Archaeological Resources

People have inhabited the Middle Columbia River region since the end of the glacial period. More than 10,000 years of prehistoric human activity in this largely arid environment have left extensive archaeological deposits along the river shores (Leonhardy and Rice 1970; Greengo 1982; Chatters 1989). Well-watered areas inland from the river show evidence of concentrated human activity (Chatters 1982, 1989; Daugherty 1952; Greene 1975; Leonhardy and Rice 1970; Rice 1980), and recent surveys indicate extensive, although dispersed, use of arid lowlands for hunting. Graves are common in various settings, and spirit quest monuments are still to be found on high, rocky summits of the mountains and buttes (Rice 1968a). Throughout most of the region, hydroelectric development, agricultural activities, and domestic and industrial construction have destroyed or covered the majority of these deposits. Amateur artifact collectors have had an immeasurable impact on what remains. Within the Hanford Site, from which the public is restricted, archaeological deposits found in the Hanford Reach of the Columbia River and on adjacent plateaus and mountains have been spared some of the disturbances that have befallen other sites. The Hanford Site is thus a de facto reserve of archaeological information of the kind and quality that has been lost elsewhere in the region.

Currently 248 prehistoric archaeological sites are recorded in the files of the Hanford Cultural Resources Laboratory. Of 48 sites included on the National Register of Historic Places (National Register), two are single sites, Hanford Island Site (45BN121) and Paris Site (45GR317), and the remainder are located in seven archaeological districts (Table 4.4-1). In addition, a draft request for Determination of Eligibility has been prepared for one traditional cultural property district (Gable Mountain/Gable Butte). Three other sites, Vernita Bridge (45BN90) and Tsulim (45BN412), and 45BN163, are considered eligible for the National Register. Archaeological sites include remains of numerous pithouse villages, various types of open campsites, and cemeteries along the river banks (Rice 1968a, 1980), spirit quest monuments (rock cairns), hunting camps, game drive complexes, and quarries in mountains and rocky bluffs (Rice 1968b), hunting/kill sites in lowland stabilized dunes, and small temporary camps near perennial sources of water located away from the river (Rice 1968b).

Many recorded sites were found during four archaeological reconnaissance projects conducted between 1926 and 1968 (Krieger 1928; Drucker 1948; Rice 1968a, 1968b). Systematic archaeological surveys conducted from the middle 1980s through 1993 are responsible for the remainder (e.g., Chatters 1989; Chatters and Cadoret 1990; Chatters and Gard 1992; Chatters et al. 1990, 1991, 1992, 1993). Little excavation has been conducted at any of the sites, and the Mid-Columbia Archaeological Society has done most of that work. They have conducted minor test excavations at several sites on the river banks and islands (Rice 1980) and a larger scale test at site 45BN157 (Den Beste and Den Beste 1976). The University of Idaho also excavated a portion of site 45BN179 (Rice 1980) and collaborated with the Mid-Columbia Archaeological Society on its other work. Test excavations have been conducted by the Hanford Cultural Resources Laboratory at the Wahluke (45GR306), Vernita Bridge (45BN90), and Tsulim (45BN412) sites and at 45BN446, 45BN423, 45BN163, 45BN432, and 45BN433; results support assessments of significance for those sites. Most of the archaeological survey and reconnaissance activity has concentrated on islands and on a strip of land less than 400 meters wide

Table 4.4-1. Archaeological districts and historic properties on the Hanford Site listed on the National Register of Historic Places (with their archaeological sites).

District/Property Name	Site(s) Included
Wooded Island A.D.[a]	45BN107 through 45BN112, 45BN168
Savage Island A.D.	45BN116 through 45BN119, 45FR257 through 45FR262
Hanford Island Site	45BN121
Hanford North A.D.	45BN124 through 45BN134, 45BN178
Locke Island A.D.	45BN137 through 45BN140, 45BN176, 45GR302 through 45GR305
Ryegrass A.D.	45BN149 through 45BN157
Paris Site	45GR317
Rattlesnake Springs A.D.	45BN170, 45BN171
Snively Canyon A.D.	45BN172, 45BN173
100-B Reactor	NA[b]

a. A.D. indicates archaeological district (this table).
b. Not applicable.

on either side of the river (Rice 1980), but this is changing because of a Hanford Cultural Resources Laboratory effort to inventory a 10 percent sample of the site by 1994. During his reconnaissance of the Hanford Site in 1968, Rice inspected portions of Gable Mountain, Gable Butte, Snively Canyon, Rattlesnake Mountain, and Rattlesnake Springs but gave little attention to other areas (Rice 1968b). He also inspected additional portions of Gable Mountain and part of Gable Butte in the late 1980s (Rice 1987). Other reconnaissance of the Basalt Waste Isolation Project Reference Repository Location (RRL) (Rice 1984) included a proposed land exchange in T22N, R27E, Section 33 (Rice 1981), and three narrow transportation and utility corridors (Ertec Northwest, Inc. 1982; Morgan 1981; Smith et al. 1977). The 100 Areas were surveyed in 1991 through 1993, revealing a large number of new archaeological sites (Chatters et al. 1992; Wright 1993). To date only about 6 percent of the Hanford Site has been surveyed. Cultural resource reviews are conducted when projects are proposed for areas that have not been previously reviewed; about 100 to 120 reviews were conducted annually through 1991; this figure rose to more than 400 reviews during 1993.

4.4.2 Native American Cultural Resources

In prehistoric and early historic times, the Hanford Reach of the Columbia River was heavily populated by Native Americans of various tribal affiliations. The Wanapum and the Chamnapum band of the Yakama tribe dwelt along the Columbia River from south of Richland upstream to Vantage (Relander 1956; Spier 1936). Some of their descendants still live nearby at Priest Rapids, and others have been incorporated into the Yakama and Umatilla reservations. Palus people, who lived on the lower Snake River, joined the Wanapum and Chamnapum to fish the Hanford Reach of the Columbia River and some inhabited the river's east bank (Relander 1956; Trafzer and Scheuerman 1986). Walla Walla and Umatilla people also made periodic visits to fish in the area. These people retain traditional secular and religious ties to the region, and many, young and old alike, have knowledge of the ceremonies and lifeways of their aboriginal culture. The Washane, or Seven Drums religion, which has ancient roots and had its start on what is now the Hanford Site, is still practiced by many people on the Yakama, Umatilla, Warm Springs, and Nez Perce reservations. Native plant and animal foods, some of which can be found on the Hanford Site, are used in the ceremonies performed by sect members.

4.4.3 Historic Archaeological Resources

The first Euro-Americans who came to this region were Lewis and Clark, who traveled along the Columbia and Snake rivers during their 1803-1806 exploration of the Louisiana Territory. They were followed by fur trappers, who also passed through on their way to more productive lands upriver and downstream and across the Columbia Basin. It was not until the 1860s that merchants set up stores, a freight depot, and the White Bluffs Ferry on the Hanford Reach. Chinese miners began to work the gravel bars for gold. Cattle ranches opened in the 1880s and farmers soon followed. Several small, thriving towns, including Hanford, White Bluffs, and Ringold, grew up along the riverbanks in the early 20th century. Other ferries were established at Wahluke and Richmond. The towns and nearly all other structures were razed after the U.S. Government acquired the land for the Hanford Nuclear Reservation in the early 1940s (Chatters 1989; Ertec Northwest, Inc. 1981; Rice 1980).

Historic archaeological sites totaling 202 and 11 other historic localities have been recorded by the Hanford Cultural Resources Laboratory on the Hanford Site. Localities include the Allard Pumping Plant at Coyote Rapids, the Hanford Irrigation Ditch, the Hanford townsite, Wahluke Ferry, the White Bluffs townsite, the Richmond Ferry, Arrowsmith townsite, a cabin at

East White Bluffs ferry landing, the White Bluffs road, the old Hanford High School, and the Cobblestone Warehouse at Riverland (Rice 1980). Archaeological sites including the East White Bluffs townsite and associated ferry landings and an assortment of trash scatters, homesteads, corrals, and dumps have been recorded by the Hanford Cultural Resources Laboratory since 1987. Ertec Northwest, Inc. was responsible for minor test excavations at some of the historic sites, including the Hanford townsite locality. In addition to the recorded sites, numerous unrecorded site areas of gold mine tailings along the river bank and the remains of homesteads, farm fields, ranches, and abandoned Army installations are scattered over the entire Hanford Site. Of these historic sites, one is included in the National Register as an historic site, and 56 are listed as archeological sites.

More recent locations are the defense reactors and associated materials processing facilities that now dominate the site. The first reactors (B, D, and F) were constructed in 1943 as part of the Manhattan Project. Plutonium for the first atomic explosion and the bomb that destroyed Nagasaki to end World War II was produced in the B Reactor. Additional reactors and processing facilities were constructed after World War II during the Cold War. All reactor containment buildings still stand, although many ancillary structures have been removed. The B Reactor has been listed on the National Register of Historic Places. A historic context for Manhattan Project facilities has been created as part of a Multiple Property Document. Until a full evaluation of all Manhattan Project buildings and facilities has been completed, statements about National Register status cannot be made.

4.4.4 200 Areas

An archaeological survey has been conducted of all undeveloped portions of the 200-East Area, and a 50 percent random sample has been conducted of undeveloped portions of the 200-West Area. The old White Bluffs freight road (see Rice 1984) crosses diagonally through the 200-West Area. The road, formerly a Native American trail, has been in continuous use since antiquity and has played a role in Euro-American immigration, development, agriculture, and Hanford Site operations. The road has been found to be eligible for listing on the National Register of Historic Places. A 100-m easement has been created to protect the road from uncontrolled disturbance. Historic buildings that have not been evaluated for National Register eligibility occur in both the 200-East and 200-West Areas.

4.5 Aesthetic and Scenic Resources

The land in the vicinity of the Hanford Site is generally flat with little relief. Rattlesnake Mountain, rising to 1060 meters (3477 feet) above mean sea level, forms the western boundary of the site. Gable Mountain and Gable Butte are the highest land forms within the site. The view toward Rattlesnake Mountain is visually pleasing, especially in the springtime when wildflowers are in bloom. Large rolling hills are located to the west and far north. The Columbia River, flowing across the northern part of the site and forming the eastern boundary, is generally considered scenic, with its contrasting blue against a background of brown basaltic rocks and desert sagebrush. The White Bluffs, steep whitish-brown bluffs adjacent to the Columbia River and above the northern boundary of the river in this region, are a striking feature of the landscape.

The potential project site (under all alternatives except No Action) is characterized by large sagebrush, desert grasses, and shrubs. Immediate views to the east include the 200-East Area facilities, views in the distant north area of reactors. Somewhat hidden by a slight rise in the land are stacks for facilities in 200-West Area to the west of the project site. To the south southwest are gravel borrow pit and radio and meteorological towers. This site is of low sensitivity in terms of aesthetic and scenic resources.

4.6 Geology

This section summarizes the geologic setting, including potential geologic hazards, at the Hanford Site. Physiography, structure, soils, and seismicity and volcanic hazards are briefly discussed. A more detailed discussion of these subjects can be found in Cushing (1992).

4.6.1 General Geology

The Hanford Site lies within the Columbia Intermontane physiographic province, bordered on the north and east by the Rocky Mountains and on the west by the Cascade Range. The dominant geologic characteristics of the Hanford Site have resulted from basaltic volcanism and ancient catastrophic flooding.

Fluvial and lacustrine processes associated with the ancestral Columbia River system, including the ancestral Snake and Yakima rivers, have been active since the late Miocene. Deposits of these rivers and lakes are represented by the Ringold Formation and indicate that deposition was almost continuous from about 10.5 million years before present until about 3.9 million years before present (DOE 1988). At some time before 900,000 years ago, a major change in regional base level resulted in fluvial incision of as much as 150 meters (500 feet). The post-Ringold erosional surface was partially filled with locally derived alluvium and fluvial sediment before and possibly between periods of Pleistocene flooding. However, in most areas of the Columbia Basin subprovince, the record of Pleistocene fluvial activity was destroyed by cataclysmic flooding. Loess (buff-colored silt) occurs in sheets that mantle much of the upland areas of the Columbia Basin subprovince.

Quaternary[a] volcanism has been limited to the extreme western margin of the Columbia Basin subprovince and is associated with the Cascade Range Province. Airfall tephra[b] from at least three Cascade volcanoes has blanketed the central Columbia Plateau since the late Pleistocene. This tephra includes material from several eruptions of Mount St. Helens before the May 1980 eruption. Other volcanoes have erupted less frequently; two closely spaced eruptions from Glacier Peak about 11,200 years ago, and the eruption of Mount Mazama about 6,600 years ago. Generally tephra layers have not exceeded more than a few centimeters in thickness, with the exception of the Mount Mazama eruption when as much as 10 centimeters (3.9 inches) of tephra fell over eastern Washington (DOE 1988).

4.6.1.1 Physiography. The Hanford Site, located within the Pasco Basin of the Columbia Plateau, is defined generally by a thick accumulation of basaltic lava flows that extend laterally from central Washington eastward into Idaho and southward into Oregon (Tallman et al. 1979).

The Hanford Site overlies the structural low point of the Pasco Basin near the confluence of the Yakima and Columbia rivers. The boundaries of the Pasco Basin are defined by anticlinal structures of basaltic rock. These structures are the Saddle Mountains to the north; the Umptanum Ridge, Yakima Ridge, and Rattlesnake Hills to the west; and the Rattlesnake

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a. Quaternary - A geologic period beginning approximately two million years ago and extending to the present.

b. Tephra - A collective term for all clastic materials ejected from a volcano and transported through the air.

Hills and a series of doubly plunging anticlines merging with the Horse Heaven Hills to the south. The terrain within the Pasco Basin is relatively flat. Its surface features were formed by catastrophic floods and have undergone little modification since, with the exception of more recently formed sand dunes (DOE 1986a).

The elevations of the alluvial plain that covers much of the site vary from 105 meters (345 feet) above mean sea level in the southeast corner to 245 meters (803 feet) in the northwest. The 200-Area plateau in the central part of the site varies in elevation from 190 to 245 meters (623 to 803 feet).

The major geologic units of the Hanford Site are (in ascending order): subbasalt rocks (inferred to be sedimentary and volcanoclastic rocks), the Columbia River Basalt Group with intercalated sediments of the Ellensburg formation, the Ringold formation, the Plio-Pleistocene unit, and the Hanford formation. Locally, sand and silt exist as surface material. A generalized stratigraphic column is shown in Figure 4.3.

Knowledge of the subbasalt rocks is limited to studies of exposures along the margin of the Columbia Plateau and to a few deep boreholes drilled in the interior of the plateau (DOE 1988). No subbasalt rocks are exposed within the central interior of the Columbia Plateau, including the Pasco Basin. Interpretation of data from wells drilled in the 1980s by Shell Oil Company in the northwestern Columbia Plateau indicates that in the central part of the Columbia Plateau the Columbia River Basalt Group is underlain predominantly by Tertiary continental sediments (Campbell 1989).

The Hanford formation lies on the eroded surface of the Plio-Pleistocene unit, on the Ringold formation, or locally on the basalt bedrock. The Hanford formation consists of catastrophic flood sediments that were deposited when ice dams in western Montana and northern Idaho were breached and massive volumes of water spilled abruptly across eastern and central Washington. The floods scoured the land surface, locally eroding the Ringold formation, the basalts, and sedimentary interbeds, leaving a network of buried channels crossing the Pasco Basin (Tallman et al. 1979). Thick sequences of sediments were deposited by several episodes of flooding with the last major flood sequence dated at about 13,000 years before the present (Myers et al. 1979).

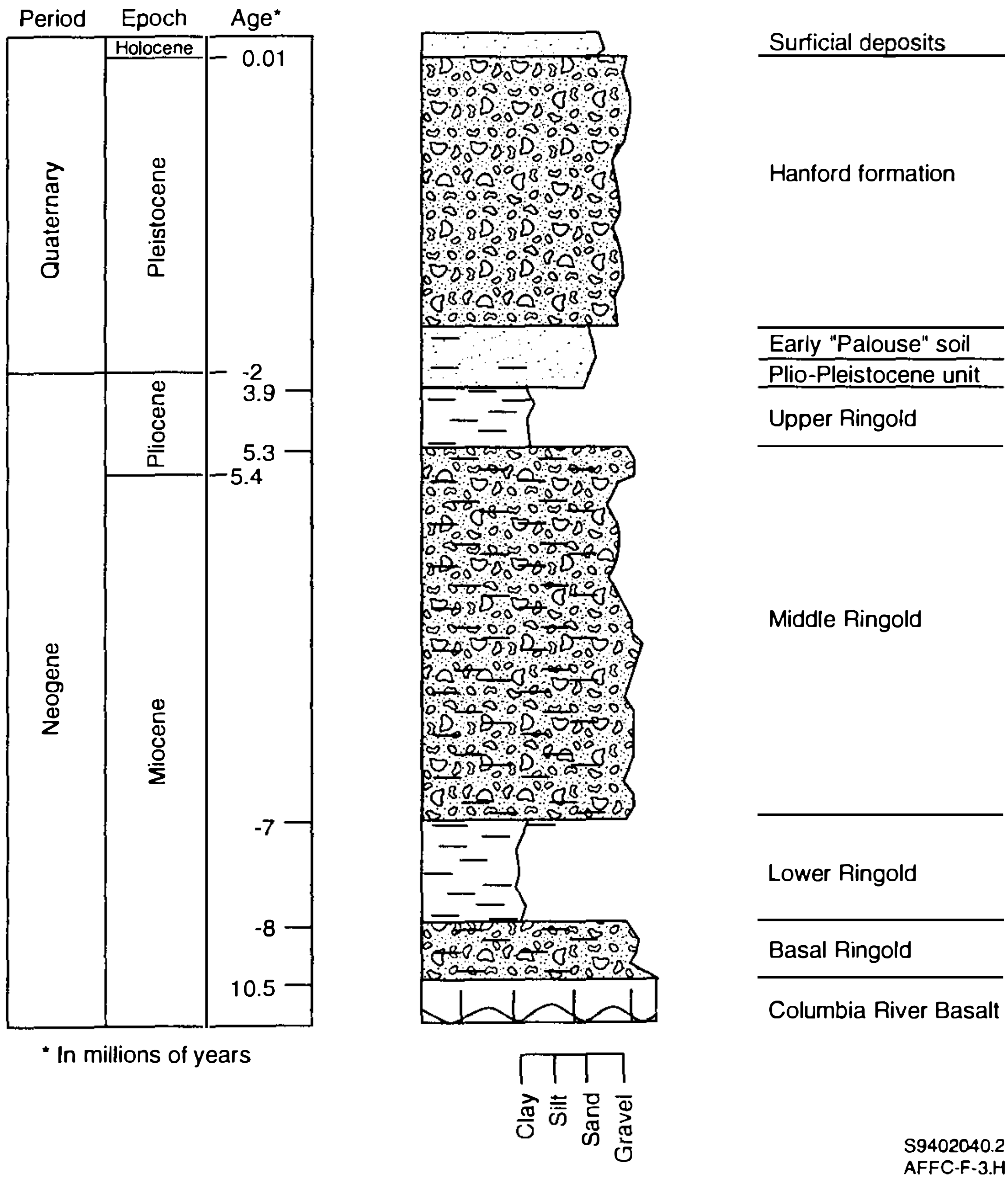

Figure 4-3. A generalized stratigraphic column of the major geologic units of the Hanford Site.

4.6.1.2 Structure. The Columbia Plateau is tectonically a part of the North American continental plate, and is separated from the Pacific and Juan de Fuca oceanic plates to the west by the Cascade Range, Puget-Willamette Lowland, and Coast Range geologic provinces. It is bounded on the north by the Okanogan Highlands, on the east by the Northern Rocky Mountains and Idaho Batholith, and on the south by the High Lava plains and Snake River plain. The tectonic history of the Columbia Plateau has included the eruption of the continental flood basalts of the Columbia River Basalt Group during the period of about 17 to 6 million years before present, as well as volcanic activity in the Cascade Range to the west (DOE 1988).

Structurally, the Columbia Plateau can be divided into three informal subprovinces: the Palouse, Blue Mountains, and Yakima Fold Belt. All but the easternmost part of the Pasco Basin is within the Yakima Fold Belt structural subprovince (DOE 1988). The Yakima Fold Belt contains four major structural elements: the Yakima Folds, Cle Elum-Wallula disturbed zone, Hog Ranch-Naneum anticline, and northwest-trending wrench faults.

The Yakima Folds are a series of continuous, narrow, asymmetric anticlines that have wavelengths between about 5 and 30 kilometers (3 to 19 miles) and amplitudes commonly less than 1 kilometers (less than 0.6 miles). The anticlinal ridges are separated by broad synclines or basins. The Yakima Folds are believed to have developed under generally north-south compression, but the origin and timing of the deformation along the fold structures are not well known (DOE 1988). Thrust or high-angle reverse faults are often found along both limbs of the anticlines, with the strike of the fault planes parallel or subparallel to the axis of the anticlines. Very little direct field evidence indicates quaternary movement along these anticlinal ridges. One of three cases of suspected Quaternary faulting is along the central Gable Mountain fault in the Pasco Basin. This fault is on the Hanford Site. It was considered by the NRC to be presumed capable, but not demonstrated to be capable for licensing purposes of the WNP plant.

The Cle Elum-Wallula disturbed zone is the central part of a larger topographic alignment called the Olympic-Wallowa lineament that extends from the northwestern edge of the Olympic Mountains to the northern edge of the Wallowa Mountains in Oregon. The Cle Elum-Wallula disturbed zone is a narrow zone about 10 kilometers (6 miles) wide that transects the Yakima Fold Belt and has been divided informally into three structural domains: a broad zone of deflected or anomalous fold and fault trends extending south of Cle Elum, Washington to Rattlesnake Mountain; a narrow belt of aligned domes and doubly plunging anticlines (called The Rattles) extending from Rattlesnake Mountain to Wallula Gap; and the Wallula fault zone, extending from Wallula Gap to the Blue Mountains. Evidence for quaternary deformation has been reported for 14 localities in or directly associated with the Cle Elum-Wallula disturbed zone. However, no evidence has been reported northwest of the Finley Quarry location (DOE 1988), about 60 kilometers (36 miles) southeast of the approximate center of the Hanford Site.

The Hog Ranch-Naneum Ridge anticline is a broad structural arch that extends from southwest of Wenatchee, Washington to the Yakima Ridge. This feature defines part of the northwestern boundary of the Pasco Basin, but little is known about the structural geology of this portion of the feature, and the southern extent of the feature is not known.

Northwest-trending wrench (strike-slip) faults have been mapped west of 120°W longitude in the Columbia Plateau (DOE 1988). The mean strike direction of the dextral wrench faults is 320°, but northeast-trending sinistral wrench faults that strike 013° are less numerous. These structures are not known to exist in the central Columbia Plateau.

Most known faults within the Hanford area are associated with anticlinal fold axes, are thrust or reverse faults although normal faults do exist, and were probably formed concurrently with the folding (DOE 1988). Existing known faults within the Hanford area include wrench (strike-slip) faults as long as 3 kilometers (1.9 miles) on Gable Mountain and the Rattlesnake-Wallula alignment, which has been interpreted as a right-lateral strike-slip fault. The faults in Central Gable Mountain are considered NRC capable by the U.S. Nuclear Regulatory Commission criteria (10 CFR 100) in that they have slightly displaced the Hanford formation gravels, but their relatively short lengths give them low seismic potential. No seismicity has been observed on or near Gable Mountain. The Rattlesnake-Wallula alignment is interpreted as possibly being capable, in part because of lack of any distinct evidence to the contrary and because this structure continues along the northwest trend of faults that appear active at Wallula Gap, some 56 kilometers (35 miles) southeast of the central part of the Hanford Site (DOE 1988).

Strike-slip faults have not been observed crosscutting the Pasco Basin. Anticlinal ridges that bound the Pasco Basin have been mapped in detail, and except for some component of dextral movement on the Rattlesnake-Wallula alignment, no strike-slip faults similar to those in the western Yakima Fold Belt have been observed (DOE 1988). Wrench (strike-slip) faults have been observed along the ridges at boundaries between geometrically coherent segments of the structures, as in the Saddle Mountains, but these faults are confined to the individual structures and formed as different geometries developed in the fold. Similar type faults have been mapped on Gable Mountain and studied in detail. These features are also interpreted as wrench (stike-slip) faults that are a response to folding.

In general, for structures within the Hanford Site area, the greatest deformation occurs in the hinge area of the anticlinal ridges and decreases with distance from that area; that is, the greatest amount of tectonic jointing and faulting occurs in the hinge zone and decreases toward the gently dipping limbs. The faults usually exhibit low dips with small displacements, may be confined to the layer in which they occur, and die out to no recognizable displacement in short lateral distances (DOE 1988).

4.6.1.3 Soils. Hajek (1966) lists and describes 15 different soil types on the Hanford Site. The soil types vary from sand to silty and sandy loam. Various classifications, including land use, are also given in Hajek (1966). The proposed SNF facility site does not contain prime or unique farmland.

Section 4.8.2.1 (Groundwater Hydrology) provides a full discussion on ranges of thickness of the various geological units/soil types across the Hanford Site (Figures 4-3 and 4-11). The surface Hanford Formation varies in thickness across the Hanford Site from approximately 15 to 100 meters (49 to 328 feet) thick (Figure 4-11). The Middle Ringold Formation varies from 10 to 100 meters (32 to 328 feet) thick. The Lower Ringold and Basal Ringold Formations only extend eastward from the western boundary of the Hanford Site approximately 11 kilometers (6.8 miles). The former is rather uniform in thickness at 20 meters (65 feet), while the latter demonstrates a maximum thickness of 40 meters (131 feet) at the far western boundary of the Hanford Site. Groundwater movement within these layers is also discussed in Section 4.8.2.1.

There is a rather thick vadose zone on the Hanford Site. However, conclusions drawn from studies conducted at several locations vary from no downward percolation of precipitation on the 200 Area Plateau, where soil texture is varied and layered with depth (all moisture penetrating the soil is removed by evaporation) to observations of downward water movement below the root zone in the 300 Area, where soils are coarse textured and where precipitation was above normal (DOE 1987).

4.6.2 Mineral Resources

Sand, gravel, and cobble deposits are ubiquitous components of the soils over the Columbia Basin in general and the Hanford Site in particular: therefore, any possible economic impact to these resources resulting from the siting of the proposed SNF facility or an access road would be considered negligible. However, because gravel pits occur near the proposed SNF facility site, from which the DOE has been extracting gravel for many uses on the Hanford Site, these deposits could have economic value.

4.6.3 Seismic and Volcanic Hazards

The following discussion briefly summarizes seismic and volcanic hazards on the Hanford Site. A more detailed discussion of seismic and volcanic hazards can be found in Cushing (1992).

4.6.3.1 Seismic Hazards. The historic record of earthquakes in the Pacific Northwest dates from about 1840. The early part of this record is based on newspaper reports of structural damage and human perception of the shaking, as classified by the Modified Mercalli Intensity scale, and is probably incomplete because the region was sparsely populated. Seismograph networks did not start providing earthquake locations and magnitudes of earthquakes in the Pacific Northwest until about 1960. A comprehensive network of seismic stations that provides accurate locating information for most earthquakes larger than magnitude 2.5 was installed in eastern Washington in 1969. A summary of the seismicity of the Pacific Northwest, a detailed review of the seismicity in the Columbia Plateau region and the Hanford Site, and a description of the seismic networks used to collect the data are provided in DOE (1988).

Large earthquakes (magnitude greater than 7 on the Richter scale) in the Pacific Northwest have occurred in the vicinity of Puget Sound, Washington, and near the Rocky Mountains in eastern Idaho and western Montana. A large earthquake of uncertain location occurred in north-central Washington in 1872. This event had an estimated maximum ranging from VIII to IX and an estimated magnitude of approximately 7. The distribution of intensities suggests a location within a broad region between Lake Chelan, Washington and the British Columbia border. Figure 4-4 shows the known faults occurring in the region.

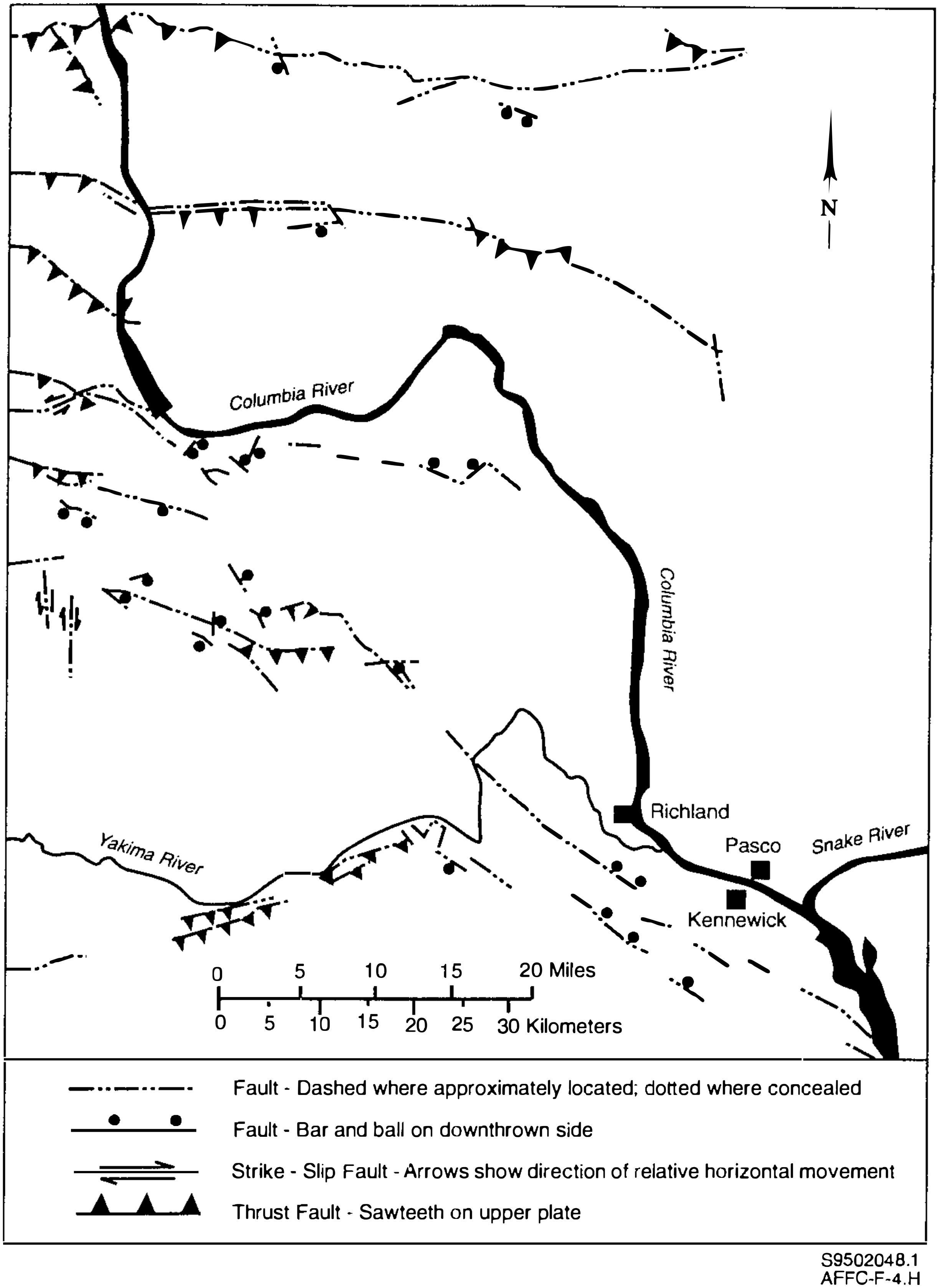

Figure 4.4. Map of the Columbia Basin region showing the known faults.

Seismicity of the Columbia Plateau, as determined by the rate of earthquakes per area and the historical magnitude of these events, is relatively low when compared to other regions of the Pacific Northwest, the Puget Sound area and western Montana/eastern Idaho. Figure 4-5 shows the locations of all earthquakes that occurred in the Columbia Plateau before 1969 with IV or larger and with a magnitude of 3 or larger. Figure 4-6 shows the locations of all earthquakes that occurred from 1969 to 1986 with magnitudes of 3 or greater. The largest known earthquake in the Columbia Plateau occurred in 1936 around Milton-Freewater, Oregon. This earthquake had a magnitude of 5.75 and a maximum of VII, and was followed by a number of aftershocks that indicate a northeast-trending fault plane. Other earthquakes with magnitudes of 5 or larger and/or intensities of VI are located along the boundaries of the Columbia Plateau in a cluster near Lake Chelan extending into the northern Cascade Range; in northern Idaho and Washington; and along the boundary between the western Columbia Plateau and the Cascade Range. Three VI earthquakes have occurred within the Columbia Plateau, including one in the Milton-Freewater region in 1921, one near Yakima, Washington in 1892, and one near Umatilla, Oregon in 1893.

In the central portion of the Columbia Plateau, the largest earthquakes near the Hanford Site are two that occurred in 1918 and 1973. These two earthquakes had magnitudes of 4.4 and an intensity of V and were located north of the Hanford Site. Earthquakes often occur in spatial and temporal clusters in the central Columbia Plateau, and are termed earthquake swarms. The region north and east of the Hanford Site is a region of concentrated earthquake swarm activity, but earthquake swarms have also occurred in several locations within the Hanford Site.

Earthquakes in a swarm tend to gradually increase and decay in frequency of events, and usually no one outstanding large event is present within the sequence. These earthquake swarms occur at shallow depths, with 75 percent of the events located at depths less than 4 kilometers (2.5 miles). Each earthquake swarm typically lasts several weeks to months, consists of several to 100 or more earthquakes, and is clustered in an area 5 to 10 kilometers (3 to 6 miles) in lateral dimension. Often, the longest dimension of the swarm area is elongated in an east-west direction. However, detailed locations of swarm earthquakes indicate that the events occur on fault planes of variable orientation, and not on a single, throughgoing fault plane.

Earthquakes in the central Columbia Plateau also occur to depths of about 30 kilometers (18 miles). These deeper earthquakes are less clustered and occur more often as single, isolated

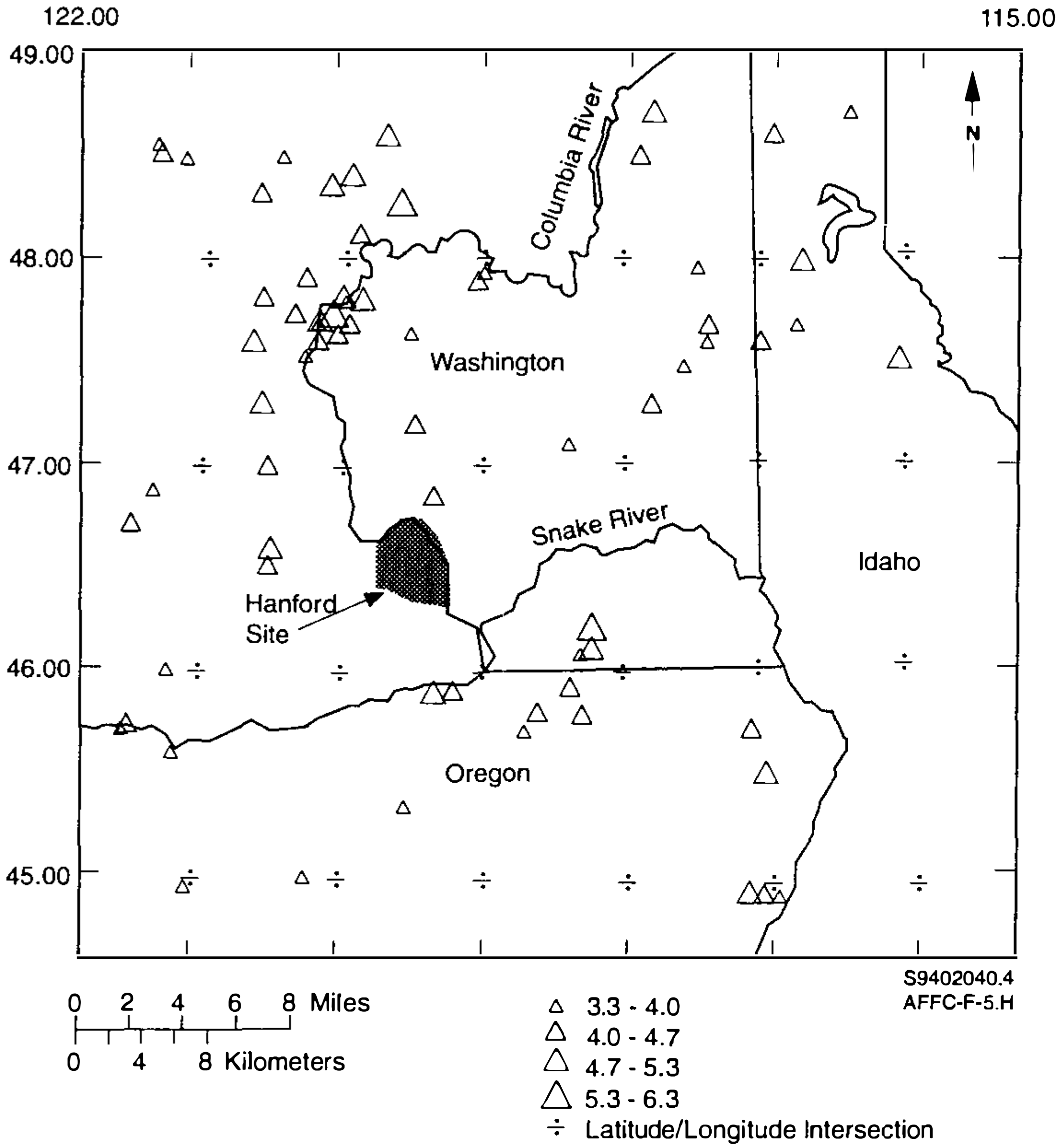

Figure 4-5. Historical seismicity of the Columbia Plateau and surrounding areas. All earthquakes between 1850 and 1969 with a Modified Mercalli Intensity of IV or larger with a magnitude of 3 or greater are shown (Rohay 1989).

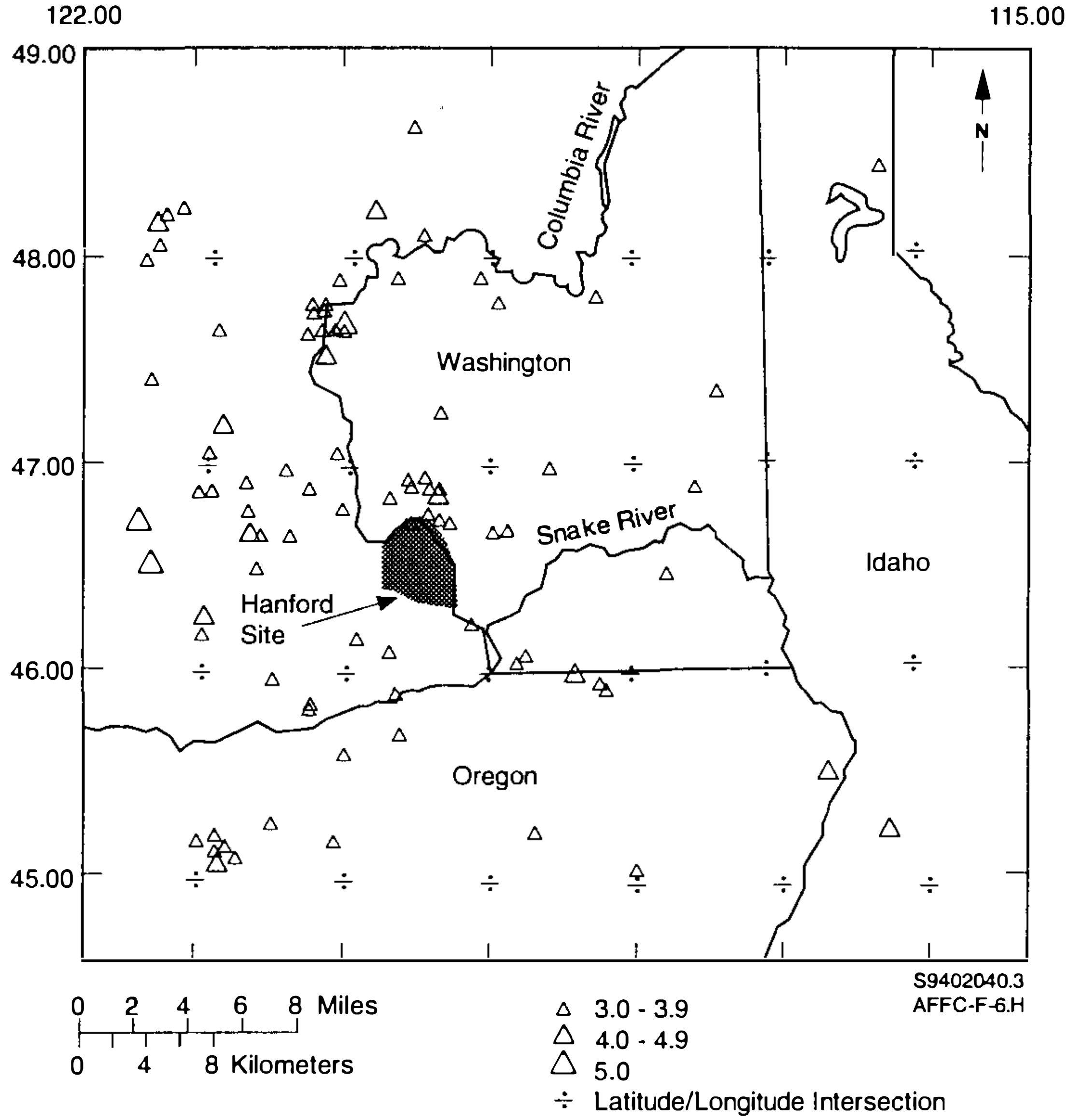

Figure 4-6. Recent seismicity of the Columbia Plateau and surrounding areas as measured by seismographs. All earthquakes between 1969 and 1986 with a Modified Mercalli Intensity of IV or larger with a magnitude of 3 or greater are shown (Rohay 1989).

events. Based on seismic refraction surveys in the region, the shallow earthquake swarms are occurring in the Columbia River Basalts, and the deeper earthquakes are occurring in crustal layers below the basalts.

The spatial pattern of seismicity in the central Columbia Plateau suggests an association of the shallow swarm activity with the east-west-oriented Saddle Mountains anticline. However, this association is complex, and the earthquakes do not delineate a throughgoing fault plane that would be consistent with the faulting observed on this structure.

Earthquake mechanisms in the central Columbia Plateau generally indicate reverse faulting on east-west planes, consistent with a north-south-directed maximum compressive stress and with the formation of the east-west-oriented anticlinal fold of the Yakima Fold Belt (Rohay 1987). However, earthquake focal mechanisms indicate faulting on a variety of fault plane orientations.

Earthquake focal mechanisms along the western margin of the Columbia Plateau also indicate north-south compression, but here the minimum compressive stress is oriented east-west, resulting in strike-slip faulting (Rohay 1987). Geologic studies indicate an increased component of strike-slip faulting in the western portion of the Yakima Fold Belt. Earthquake focal mechanisms in the Milton-Freewater region to the southeast indicate a different stress field, one with maximum compression directed east-west instead of north-south.

Estimates for the earthquake potential of structures and zones in the central Columbia Plateau have been developed during the licensing of nuclear power plants at the Hanford Site. In reviewing the operating license application for a Washington Public Power Supply System project, the Nuclear Regulatory Commission (NRC 1982) concluded that four earthquake sources should be considered for the purpose of seismic design: the Rattlesnake-Wallula alignment, Gable Mountain, a floating earthquake in the tectonic province, and a swarm area.

For the Rattlesnake-Wallula alignment, which passes along the southwest boundary of the Hanford Site, the estimated maximum magnitude is 6.5, and for Gable Mountain, an east- west structure that passes through the northern portion of the Hanford Site, the estimated maximum magnitude is 5.0. These estimates were based upon the inferred sense of slip, the fault length, or the fault area. The floating earthquake for the tectonic province was developed from the largest event located in the Columbia Plateau, the magnitude 5.75 Milton-Freewater earthquake.

The maximum swarm earthquake for the purpose of seismic design was a magnitude 4.0 event. Figures 4-7 through 4-11 demonstrate the ranges of frequencies versus the acceleration across the Hanford Site (Geomatrix Consultants, Inc. 1993).

The seismic design is based upon a Safe-Shutdown Earthquake of 0.25 gravity (g; acceleration). The potential earthquake risk associated with the Gable Mountain structure dominated the risks associated with other potential sources that were considered. For DOE site comparison purposes, a maximum horizontal ground surface acceleration of 0.17-0.20g at the Hanford Site is estimated to result from an earthquake that could occur once every 2,000 years (DOE 1994c). The seismic hazard information presented in this EIS is for general seismic hazard comparisons across DOE sites. Potential seismic hazards for existing and new facilities could be evaluated on a facility specific basis consistent with DOE orders and standards and site specific procedures.

4.6.3.2 Volcanic Hazards. Several major volcanoes are located in the Cascade Range west of the Hanford Site. The nearest volcano, Mount Adams, is about 165 kilometers (102 miles) from the Hanford Site, and the most active is Mount St. Helens, approximately 220 kilometers (136 miles) west-southwest from Hanford.

A period of renewed volcanic activity at Mount St. Helens began in March 1980 and climaxed in a major eruption on May 18, 1980. This eruption resulted in about 1 millimeter (0.039 inches) of ash fall over a 9-hour period at the Hanford Site, which was near the southern edge of the ash dispersal plume. Smaller eruptions of steam and ash occurred through October 1980, but none of these deposited measurable amounts of ash at the site. Because of their close proximity, the volcanic mountains of the Cascades are the principal volcanic hazard at Hanford.

The major concern is how ash fall might affect the operation of communications equipment and electronic devices, as well as the movement of truck and automobile traffic in and out the project site area.

4.7 Air Resources

This section addresses the general air resources at the Hanford Site and surrounding region. Included in this section are discussions on climate and meteorology, ambient air quality, and atmospheric dispersion.

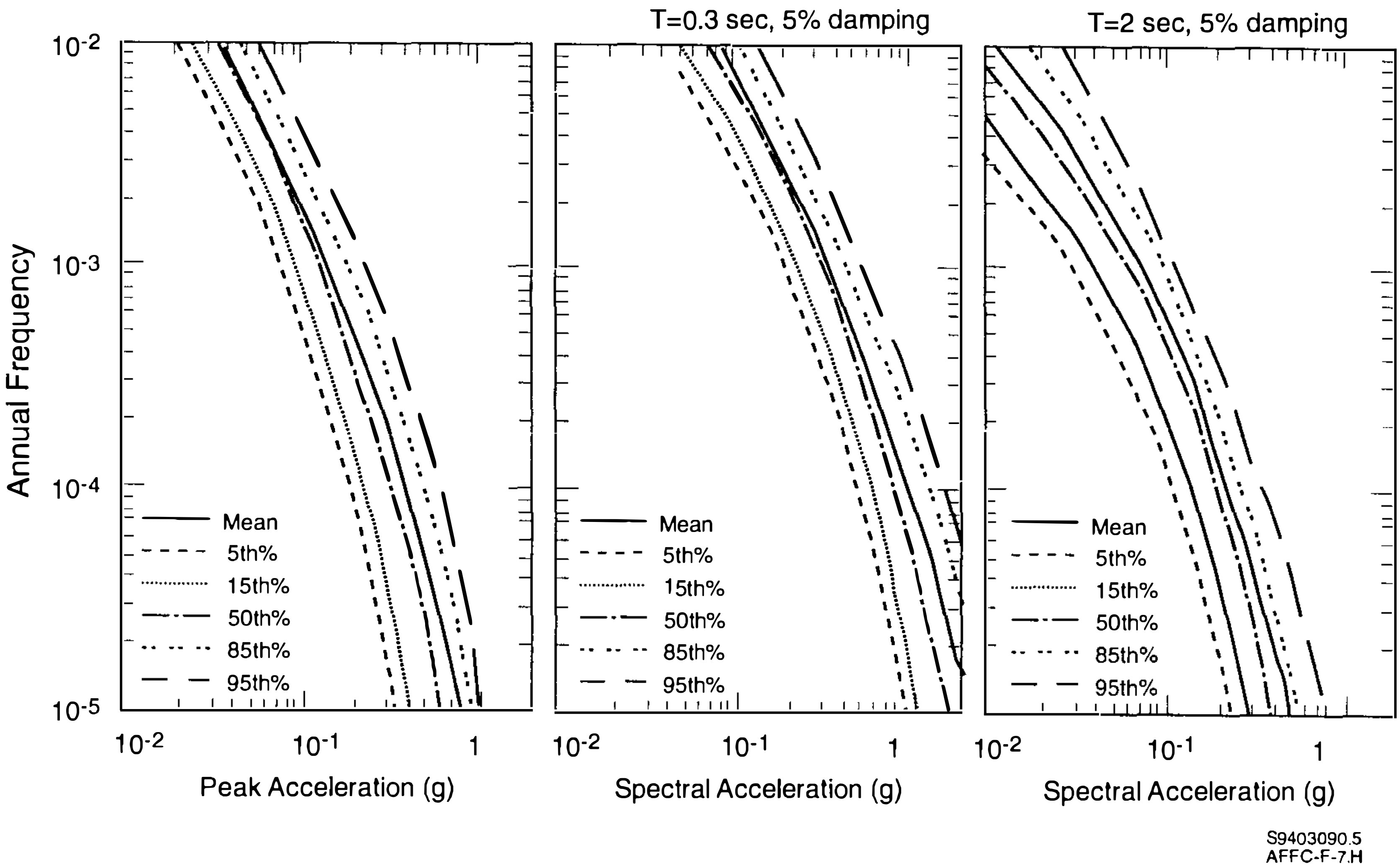

Figure 4-7. Computed mean and 5th to 95th percentile hazard curves for the 200-West Area of the Hanford Site. Shown are results for peak horizontal acceleration and 5 percent-damped spectral acceleration at 0.3 and 2.0 seconds (Geomatrix Consultants, Inc. 1993).

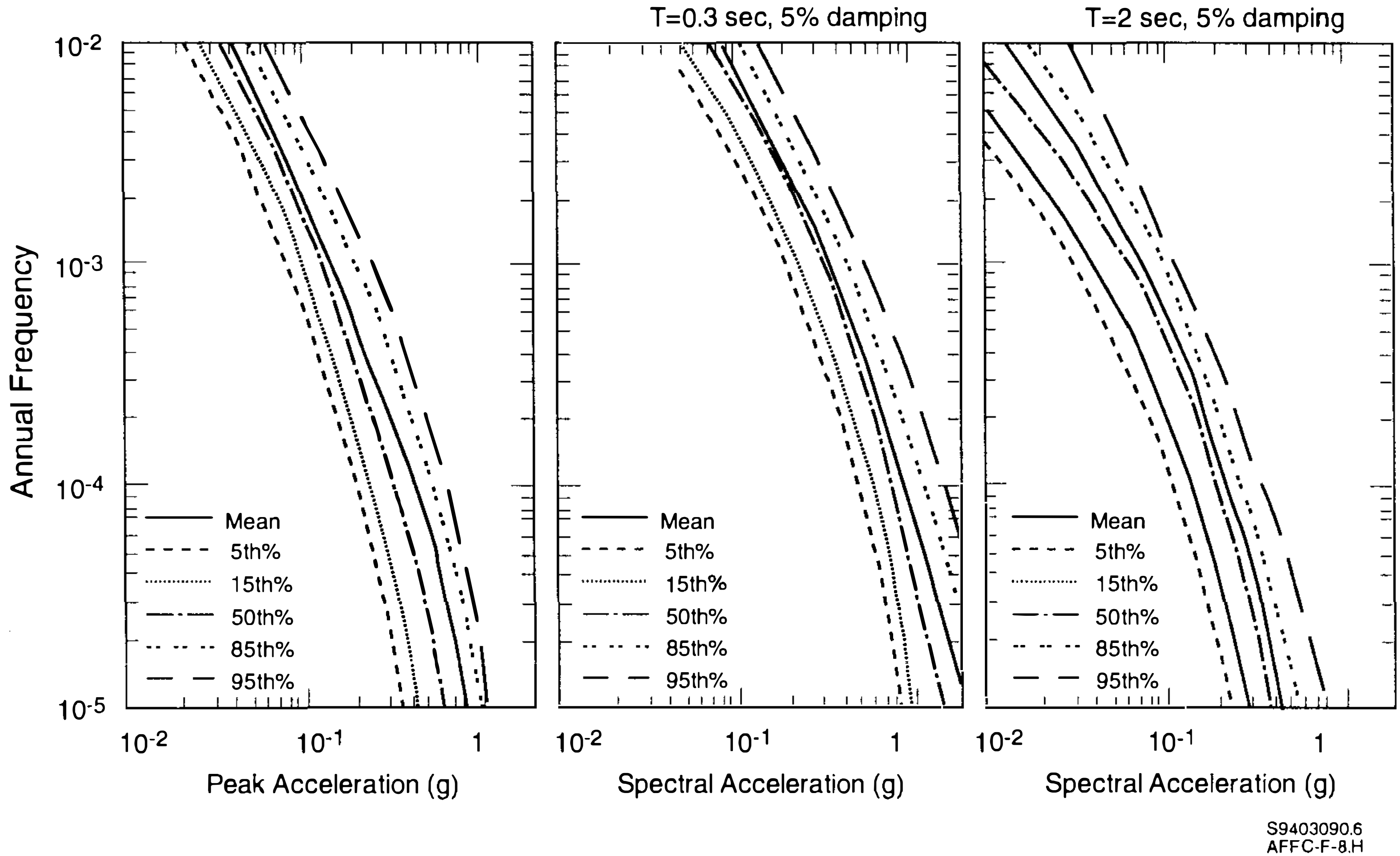

Figure 4-8. Computed mean and 5th to 95th percentile hazard curves for the 200-East Area of the Hanford Site. Shown are results for peak horizontal acceleration and five percent-damped spectral acceleration at 0.3 and 2.0 seconds (Geomatrix Consultants, Inc. 1993).

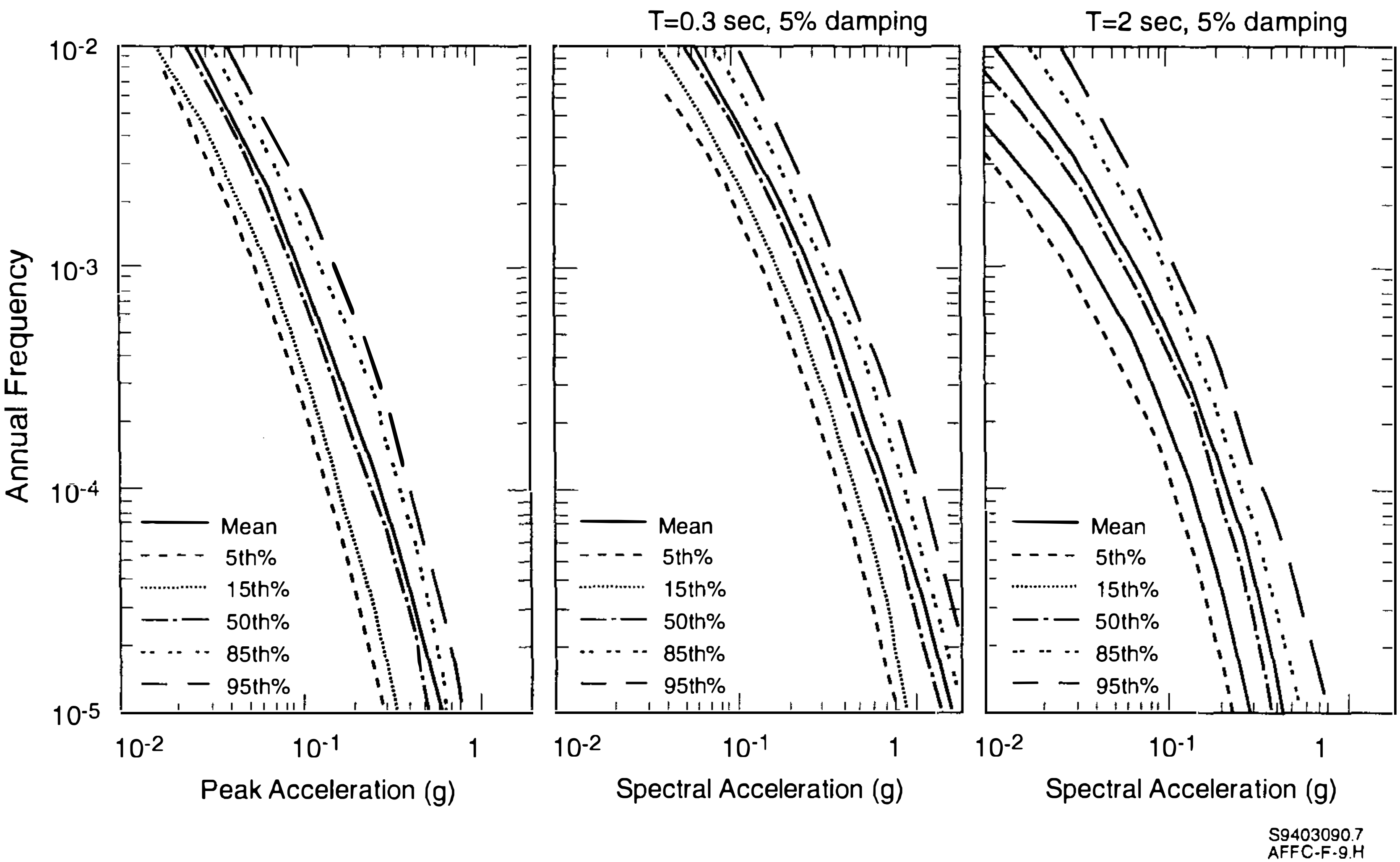

Figure 4-9. Computed mean and 5th to 95th percentile hazard curves for the 300 Area of the Hanford Site. Shown are results for peak horizontal acceleration and five percent-damped spectral acceleration at 0.3 and 2.0 seconds (Geomatrix Consultants, Inc. 1993).

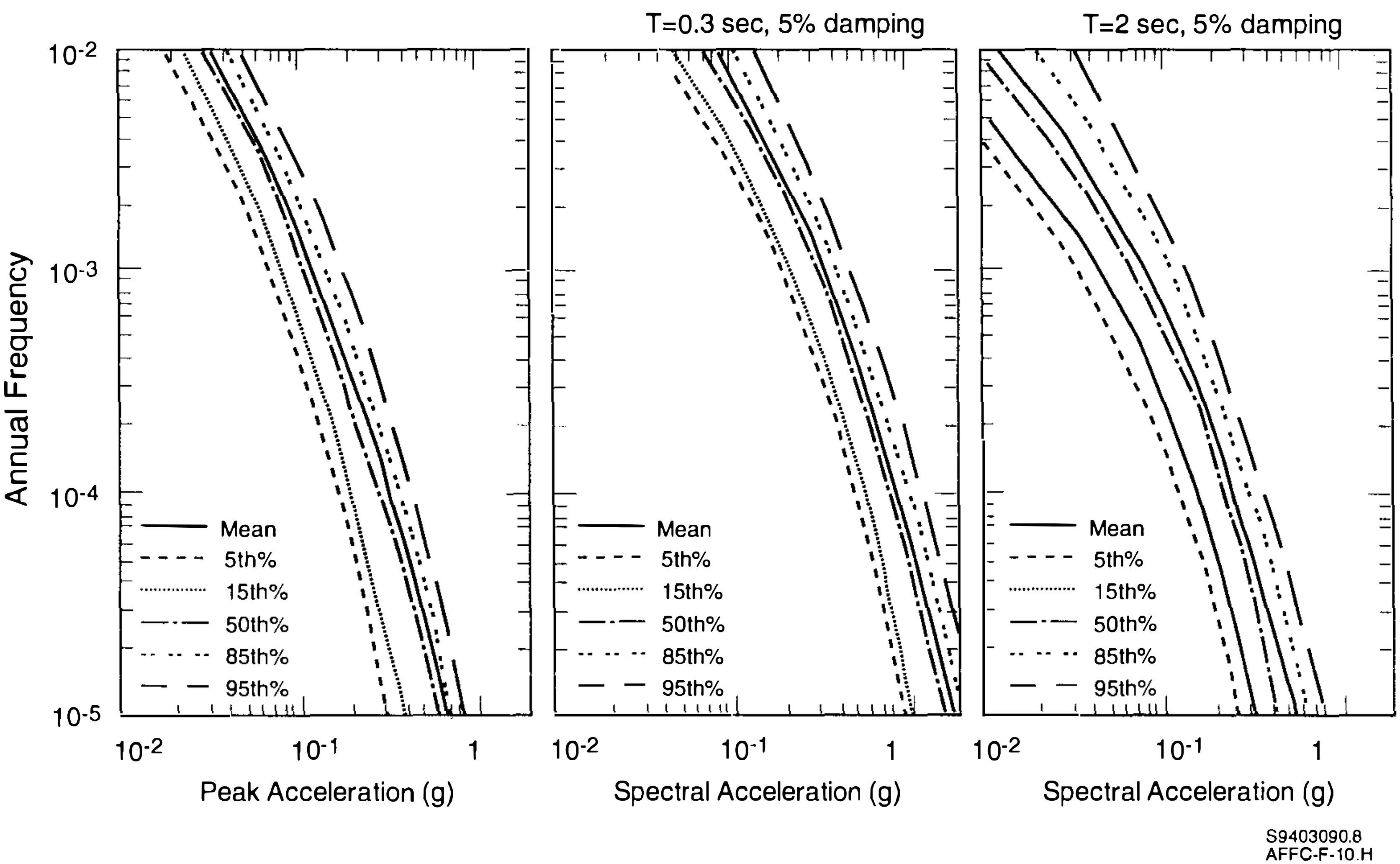

Figure 4-10. Computed mean and 5th to 95th percentile hazard curves for the 400 Area of the Hanford Site. Shown are results for peak horizontal acceleration and five percent-damped spectral acceleration at 0.3 and 2.0 seconds (Geomatrix Consultants, Inc. 1993).

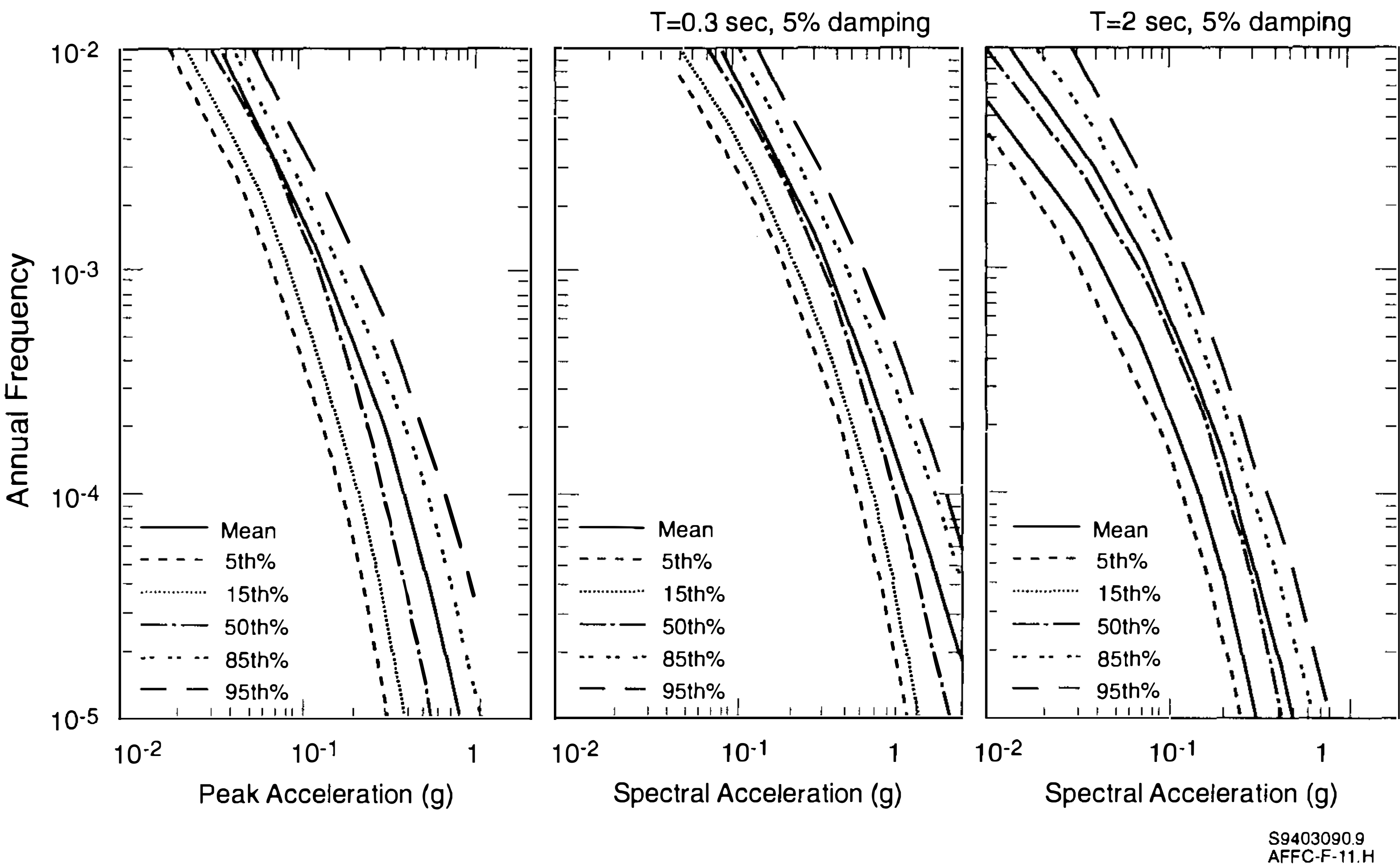

Figure 4-11. Computed mean and 5th to 95th percentile hazard curves for the 100-K Area of the Hanford Site. Shown are results for peak horizontal acceleration and five percent-damped spectral acceleration at 0.3 and 2.0 seconds (Geomatrix Consultants, Inc. 1993).

4.7.1 Climate and Meteorology

The climate of the Hanford Site, located in southcentral Washington State, can be classified as mid-latitude semiarid or mid-latitude desert, depending on the climatological classification scheme used. Summers are warm and dry with abundant sunshine. Large diurnal temperature variations result from intense solar heating during the day and radiational cooling at night. Daytime high temperatures in June, July, and August periodically exceed 38°C (100°F). Winters are cool with occasional precipitation. Outbreaks of cold air associated with modified arctic air masses can reach the area and cause temperatures to drop below -18°C (0°F). Overcast skies and fog occur periodically (Stone et al. 1983).

Topographic features have a significant impact on the climate of the Hanford Site. All air masses that reach the region undergo some modification resulting from their passage over the complex topography of the Pacific Northwest. The climate of the region is strongly influenced by the Pacific Ocean and the Cascade Range to the west. The relatively low annual average rainfall of 16.1 centimeters (6.3 inches) at the Hanford Meteorological Station is caused largely by the rain shadow created by the Cascade Range. These mountains limit much of the maritime influence of the Pacific Ocean, resulting in a more continental-type climate than would exist if the mountains were not present. Maritime influences are experienced in the region during the passage of frontal systems and as a result of movement through gaps in the Cascade Range (such as the Columbia River Gorge).

The Rocky Mountains to the east and the north also influence the climate of the region. These mountains play a key role in protecting the region from the more severe winter storms and the extremely low temperatures associated with the modified arctic air masses that move southward through Canada. Local and regional topographical features, such as the Yakima Ridge and the Rattlesnake Hills, also impact meteorological conditions across the Hanford Site (Glantz and Perrault 1991). In particular, these features have a significant impact on wind directions, wind speeds, and precipitation levels.

Climatological data are collected for the Hanford Site at the Hanford Meteorological Station. The station is located between the 200-West and 200-East Areas and is in close proximity to the proposed project site. Data have been collected at this location since 1945 and are summarized in Stone et al. (1983). Beginning in the early 1980s, data have also been collected at a series of automated monitoring sites located throughout the Hanford Site and the surrounding region (Glantz et al. 1990). This Hanford Meteorological Monitoring Network is described in detail in Glantz and Islam (1988).

4.7.1.1 Wind. Prevailing wind directions on the 200-Area plateau are from the northwest in all months of the year. Secondary maxima occur for southwesterly winds. Summaries of wind direction indicate that winds from the northwest quadrant occur most often during the winter and summer. During the spring and fall, the frequency of southwesterly winds increases with a corresponding decrease in northwest flow. Winds blowing from other directions (for instance, the northeast) display minimal variation from month to month. Monthly average wind speeds are lowest during the winter months, averaging 2.8 to 3.1 meters per second (6.2 to 6.8 miles per hour), and highest during the summer, averaging 3.9 to 4.4 meters per second (8.7 to 9.9 miles per hour). Summertime drainage winds are generally northwesterly and can frequently gust to 14 meters per second (31 miles per hour). A wind rose for the Hanford Site is shown in Figure 4-12.

4.7.1.2 Temperature and Humidity. Eight separate temperature measurements are made at the 122-meter (400-foot) tower at the Hanford Meteorological Station. As of May 1987, temperatures are also measured at the 2-meter (6.6-foot) level on the twenty-two 9.1-meter (30-foot) towers located on and around the Hanford Site. The three 61-meter (200-foot) towers have temperature-measuring instrumentation at the 2-, 9.8-, and 61-meter (6.6-, 32-, and 200-foot) levels. The temperature data from the 9.1- and 61-meter (30- and 200-foot) towers are telemetered to the Hanford Meteorological Station.

Diurnal and monthly averages and extremes of temperature, dew point, and humidity are contained in Stone et al. (1983). Ranges of daily maximum and minimum temperatures vary from normal maxima of 2°C (36°F) in early January to 35°C (95°F) in late July. On the average, 55 days during the summer months have maximum temperatures greater than or equal to 32°C (90°F), and 13 days have maxima greater than or equal to 38°C (100°F). From mid-November through mid-March, minimum temperatures average less than or equal to 0°C (32°F), with the minima in early January averaging -6°C (21°F). During the winter, on average, four days have minimum temperatures less than or equal to -18°C (0°F); however, only about one winter in two experiences such temperatures. The record maximum temperature is 46°C (115°F), and the record minimum temperature is -33°C (-27°F). For the period 1912 through 1980, the average monthly temperatures ranged from a low of -1.5°C (29°F) in January to a high of 24.7°C (77°F)

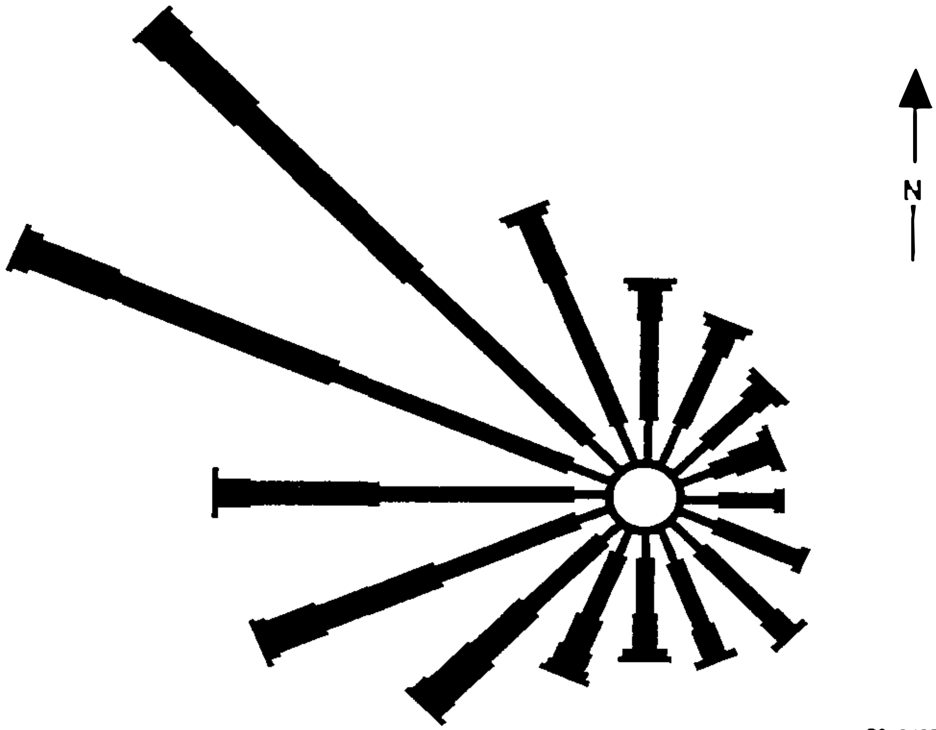

Figure 4-12. Wind rose for the Hanford Site using data collected from January 1982 to December 1989 (Glantz et al. 1990). The direction of each of the petals of the wind rose indicates the wind direction, and the petal length is representative of the percentage of time the wind was from that direction. Petal thickness represents measured wind-speed category. The velocity categories, from thinnest line (near the center of the rose) to thickest line (near the edge of the rose), are 0.4-1.3 meters per second (1-3 miles per hour), 1.8-3.1 meters per second (4-7 miles per hour), 3.6-5.4 meters per second (8-12 miles per hour), 5.8-8.0 meters per second (13-18 miles per hour), 8.5-10.7 meters per second (19-24 miles per hour), 11.2-13.9 meters per second (25-31 miles per hour), respectively.

in July. During the winter, the highest monthly average temperature at the Hanford Meteorological Station was 7°C (45°F), and the record lowest was -5.9°C (21°F), both occurring during February. During the summer, the record highest monthly average temperature was 27.9°C (82°F, in July), and the record lowest was 17.2°C (63°F, in June).

Relative humidity/dew point temperature measurements are made at the Hanford Meteorological Station and at the three 61-meter (200-foot) tower locations. The annual average relative humidity at the Hanford Meteorological Station is 54 percent. It is highest during the winter months, averaging about 75 percent, and lowest during the summer, averaging about 35 percent. Wet bulb temperatures greater than 24°C (75°F) had not been observed at the Hanford Meteorological Station before 1975; however, on July 8, 9, and 10 of that year, seven hourly observations indicated wet bulb temperatures greater than or equal to 24°C (75°F). Fog reduces the visibility to 6 miles during an average of 42 days each year and to less than 0.25 mile during an average of 25 days per year.

4.7.1.3 Precipitation. The average annual precipitation at the Hanford Meteorological Station is 16.1 centimeters (6.3 inches). Most of the precipitation occurs during the winter with nearly half of the annual amount occurring in the months of November through February. Days with greater then 1.3 centimeters (0.5 inches) precipitation occur less than 1 percent of the year. A rainfall intensity of at least 1.3 centimeters per hour (0.5 inches per hour) persisting for 1 hour has only a 10 percent probability of occurring in any given year. A rainfall intensity of at least 2.5 centimeters per hour (1 inch per hour) has only a 0.2 percent probability of occurring in any given year. Winter monthly average snowfall ranges from 0.8 centimeters (0.3 inches) in March to 13.5 centimeters (5.3 inches) in January. The record snowfall of 53 centimeters (21 inches) occurred in December 1992. During the months of December, January, and February, snowfall accounts for about 38 percent of all precipitation.

4.7.1.4 Severe Weather. A discussion of severe weather may include a variety of meteorological events, including, but not limited to, severe winds, dust and blowing dust, hail, fog, glaze, ash falls, extreme temperatures, temperature inversions, and blowing and drifting snow. These are described in detail in Stone et al. (1983). For many facilities, estimates of severe winds are of particular concern. The Hanford Meteorological Station's climatological summary and the National Severe Storms Forecast Center's database list only 24 separate tornado occurrences within 160 kilometers (100 miles) of the Hanford Site from 1916 to 1992 (Cushing 1992). Only one of these tornadoes was observed within the boundaries of the Hanford Site (on its extreme western edge), and no damage resulted. The estimated probability of a tornado striking a point at Hanford is 9.6×10^{-6} per year (Cushing 1992). Because tornadoes are infrequent and generally small in the Pacific Northwest (and hurricanes do not reach this area), risks from severe winds are generally associated with thunderstorms or the passage of strong cold fronts. The greatest peak wind gust recorded at 15 meters (50 feet) above ground level at the Hanford Meteorology Station was 36 meters per second (80 miles per hour). Projections on the return periods for peak gusts exceeding a specified speed are given in Stone et al. (1983). Extrapolations based on 35 years of observations indicate a return period of about 200 years for a peak gust in excess of 40 meters per second (90 miles per hour) at 15 meters (50 feet) above ground level.

4.7.1.5 Atmospheric Stability. The transport and diffusion of airborne pollutants is dependent on the horizontal and vertical distribution of temperature, moisture, and wind velocity in the atmosphere. Greater amounts of turbulence or mixing in an atmospheric layer lead to greater rates of diffusion. The highest rates of diffusion are found in thermally unstable layers, moderate rates of diffusion are found in neutral layers, and the lowest rates of diffusion are found in thermally stable layers. There are a number of methods for estimating the "stability" of the atmosphere. Using a method based on the vertical temperature gradient (NRC 1980) and measurements made at the Hanford Meteorology Station, thermally unstable conditions are estimated to occur an average of about 25% of the time, neutral conditions about 31% of the time, and thermally stable conditions about 44% of the time. Detailed information on Hanford's atmospheric stability and associated wind conditions are presented in Glantz et al. (1990).

4.7.2 Nonradiological Air Quality

National ambient air quality standards (NAAQS) have been set by the EPA as mandated in the 1970 Clean Air Act. Ambient air is that portion of the atmosphere, external to buildings, to which the general public has access. For DOE facilities, this is interpreted to mean the site boundary or other publicly accessible location, e.g., highways on the site. The standards define levels of air quality that are necessary, with an adequate margin of safety, to protect the public health (primary standards) and the public welfare (secondary standards). Standards exist for sulfur oxides (measured as sulfur dioxide), nitrogen dioxide, carbon monoxide, particles with an aerodynamic diameter less than or equal to 10 micrometers (PM_{10}), lead, and ozone. The standards specify the maximum pollutant concentrations and frequencies of occurrence that are allowed for specific averaging periods (that is, the concentration of carbon monoxide when averaged over 1 hour is allowed to exceed 40 milligrams per cubic meter only once per year). The averaging periods vary from 1 hour to 1 year, depending on the pollutant.

In addition to ambient air quality standards, the EPA has established standards for the Prevention of Significant Deterioration (PSD) of air quality. The PSD standards differ from the NAAQS in that the NAAQS provide maximum allowable concentrations of pollutants, while PSDs provide maximum allowable increases in concentrations of pollutants for areas already in compliance with NAAQS. Prevention of Significant Deterioration standards are expressed as allowable increments in atmospheric concentrations of specific pollutants (nitrogen dioxide, sulfur dioxide, and PM_{10}) (40 CFR 52.21, "Prevention of Significant Deterioration of Air Quality"). Different PSD standards exist for Class 1 areas (where degradation of ambient air quality is to be severely restricted), and Class II areas (where moderate degradation of air quality is allowed) (Wark and Warner 1981). The PSD standards are presented in Table 4.7-1. The nitrogen oxide emissions from the Plutonium and Uranium Recovery through EXtraction (PUREX) plant and the Uranium Oxide (UO_3) plant are permitted by the EPA under the PSD program (Cushing 1992).

State and local governments have the authority to impose standards for ambient air quality that are stricter than the national standards. Washington State has established more stringent standards for sulfur dioxide. In addition, Washington has established standards for volatile organic compounds, arsenic, fluoride, total suspended particulates, and other pollutants that are not covered by national standards. The state standards for carbon monoxide and nitrogen dioxide are identical to the national standards. At the local level, the Benton-Franklin Counties Clean Air Authority has the authority to establish more stringent air standards, but has not done so. Table 4.7-2 summarizes Washington State standards, and background and ambient concentrations for Hanford.

4.7.2.1 Background Air Quality. The closest Class I areas to the Hanford Site are Mount Rainier National Park, located approximately 160 kilometers (100 miles) west of the site; Goat Rocks Wilderness Area, located approximately 145 kilometers (90 miles) west of the site;

Table 4.7-1. Maximum allowable increases for prevention of significant deterioration of air quality[a].

Pollutant	Averaging Time	Class I	Class II
Particulate matter[b] (PM_{10})			
	annual	4	17
	24 hours	8	30
Sulfur dioxide			
	annual	2	20
	24 hours	5	91
	3 hours	25	512
Nitrogen dioxide			
	annual	2.5	25

a. Source: 40 CFR 52.21.
b. Particulate matter is defined as suspended particulates with an aerodynamic diameter less than 10 micrometers.

Table 4.7-2. Washington State ambient air quality standards applicable to Hanford, maximum background concentration, background as percent of standard, ambient baseline (1995), ambient baseline as percent of standard, and ambient baseline plus background as percent of standard (standards and concentrations are in microgram per cubic meter).[a]

Pollutant	Averaging Time	Washington State Standard	Maximum Background Concentration	Background as Percent of Standard	Ambient Baseline (effective 1995)	Ambient Baseline as percent of Standard	Ambient Baseline and Background as percent of standard
Sulfur dioxide	annual	52	0.5	1	2	4	5
	24 hour	260	6	2	19	7	10
	1 hour	1,018	49	5	127	12	17
	1 hour	655[b]	49	7	127	19	27
Particulate matter							
TSP[c]	annual	60	56	93	0	0	93
	24 hour	150	356	237	6	4	241
PM	annual	50[d]	26[e]	52	0	0	52
	24 hour	150	596[e]	397	3	2	397
Carbon monoxide	8 hour	10,000	6,500	65	3	0	65
	1 hour	40,000	11,800	30	10	0	30
Ozone	1 hour	235	not estimated	not estimated	not estimated	not estimated	not estimated
Nitrogen dioxide	annual	100	36	36	3	3	39
Lead	annual	1.5	not estimated	not estimated	not estimated	not estimated	not estimated

a. Source: Air Quality Impact Analysis in Support of the New Production Reactor Environmental Impact Statement.
b. The standard is not to be exceeded more than twice in any seven consecutive days.
c. The TSP standards have been replaced by the PM_{10} standards, but the former are serving as interim standards.
d. Arithmetic mean of the quarterly arithmetic means for the four calendar quarters of the year.
e. Maximum concentrations were measured in 1992 at Columbia Center in Kennewick. This value includes background concentration and site concentrations.

Mount Adams Wilderness Area, located approximately 150 kilometers (95 miles) southwest of the site; and Alpine Lakes Wilderness Area, located approximately 175 kilometers (110 miles) northwest of the site.

Air quality in the Hanford region is well within the state and federal standards for criteria pollutants, except that short-term particulate concentrations occasionally exceed the 24-hour PM_{10} standard (Table 4.7-2). Concentrations of toxic chemicals, as listed in 40 CFR Part 60.01, are not available for the Hanford Site. Because the highest concentrations of airborne particulate material are generally a result of natural events, the area has not been designated non-attainment[a] with respect to the PM_{10} standard. However, the local clean air authority is currently completing discussions with EPA and the Department of Ecology regarding plans to conduct additional evaluations of potential sources and mitigation measures, if any, that might be implemented to reduce the short-term particulate loading.

Particulate concentrations can reach relatively high levels in eastern Washington because of exceptional natural events (dust storms, volcanic eruptions, and large brushfires) that occur in the region. Washington ambient air quality standards do not consider rural fugitive dust from exceptional natural events when estimating the maximum background concentrations of particulate in the area east of the Cascade Mountain crest. Similarly, the EPA also exempts the rural fugitive dust component of background concentrations when considering permit applications and enforcement of air quality standards (Cushing 1992).

4.7.2.2 Source Emissions. Emissions inventories for permitted pollution sources in Benton, Franklin, and Walla Walla counties are routinely compiled by the Tri-County Air Pollution Control Board. The annual emission rates for stationary sources within the Hanford Site boundaries were reported to the Washington State Department of Ecology by the U.S. Department of Energy and are provided in Table 4.7-3.

The EPA's ISC/ST model was used for baseline modeling of stationary sources projected to be in operation in 1995 (Hadley 1991). Projected baseline conditions (presented in Table 4.7-2) are estimated to be well below any current national or state standards (Hadley 1991).

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a. An attainment area is an area where measured concentrations of a pollutant are below the primary and secondary National Ambient Air Quality Standards (NAAQS).

Table 4.7-3. Emission rates (tons per year) for stationary emission sources within the Hanford Site for 1992[a].

Source	Operation (hours per year)	TSP	PM_{10}	Sulfur Dioxide	Nitrogen Oxides	Volatile Organic Compounds	Carbon Monoxide
300 Area Boiler #2	6384	9	8	110	22	0	2
300 Area Boiler #6	8760	4	3	48	10	0	1
200-East Boiler	8760	3	1	200	58	1	49
200-West Boiler	8760	4	1	260	75	1	62
200-East, 200-West Fugitive Coal	8760	107	54	0	0	0	0
300 Area Temporary Boiler	8760	9	8	120	24	0	2
Fugitive Emissions, 200-E	8760	1	0	0	0	0	0

a. Source: Cushing in preparation.

4.7.2.3 Nonradiological Air Quality Monitoring.

4.7.2.3.1 Onsite Monitoring—The most recent monitoring data available were obtained in 1992. Details of the monitoring program are described in Woodruff and Hanf (1993). The only onsite air quality monitoring conducted during 1991 was for nitrogen oxides. These oxides were sampled at three locations on the Hanford Site with a bubbler assembly operated to collect 24-hour integrated samples. The highest annual average concentration was <0.006 parts per million by volume, well below the applicable federal and Washington State annual ambient standard of 0.05 parts per million by volume (Cushing 1992). Monitoring of total suspended solids was discontinued in early 1988 when the Basalt Waste Isolation Project, for which those measurements were required, was concluded. In 1992 sampling was done at Rattlesnake Springs (near the southwestern edge of the site) for polychlorinated biphenyls (PCBs) and volatile organic compounds. Levels of PCB concentrations were found to be ≤ 0.27 to ≤ 0.29 nanogram per cubic meter (Woodruff and Hanf 1993). These values are well below the EPA limit of 1 nanogram per cubic meter. The volatile organic compounds tested for were halogenated alkanes and alkenes, benzene, and alkylbenzenes. All volatile organic compound concentrations were well below the occupational maximum allowable concentrations of air contaminants.

4.7.2.3.2 Offsite Monitoring—During the past 10 years, carbon monoxide, sulfur dioxide, and nitrogen dioxide have been monitored periodically in communities and commercial areas southeast of Hanford. These urban measurements are typically used to estimate the maximum background pollutant concentrations for the Hanford Site because of a lack of specific onsite monitoring. Because these measurements were made in the vicinity of local sources of pollution, they will overestimate maximum background concentrations for the Hanford Site or at the site boundaries.

The only offsite monitoring in the vicinity of the Hanford Site in 1990 was conducted by the Washington Department of Ecology for particulates (WDOE 1991). Total suspended particulate (TSP) monitoring at Tri-Cities locations was discontinued in early 1989. Monitoring at the remaining two locations, Sunnyside and Wallula, continued during 1990. The annual geometric means of measurements at Sunnyside and Wallula for 1990 were 71 micrograms per cubic meter and 80 micrograms per cubic meter, respectively; both of these values exceeded the Washington State annual standard of 60 micrograms per cubic meter. The Washington State 24-hour standard, 150 micrograms per cubic meter, was exceeded six times during the year at Sunnyside and seven times at Wallula (Cushing 1992).

Particulate matter (PM_{10}) was also monitored at three locations: Columbia Center in Kennewick, Walla Walla Fire Station, and Wallula. During 1992, the 24-hour PM_{10} standard adopted by Washington State, 150 micrograms per cubic meter, was exceeded two times at the Columbia Center monitoring location. The maximum 24-hour concentration at Columbia Center was 596 micrograms per cubic meter. The maximum 24-hour concentration at the Walla Walla Fire Station was 67 micrograms per cubic meter. The maximum 24-hour concentration at Wallula was 124 micrograms per cubic meter. None of the sites exceeded the annual primary standard, 50 micrograms per cubic meter (Cushing in preparation). As noted previously, the Benton-Franklin counties area has not been designated nonattainment with respect to PM_{10} standards because the particulate concentrations result from natural events.

4.7.2.4 Summary of Nonradiological Air Quality. The Hanford Site is currently considered an attainment area for criteria pollutants. However, PM_{10} concentrations are high enough that the designation may change. There are no Class I areas close enough to the site to be affected by emissions at Hanford. Carbon monoxide concentrations are at 65 percent of the allowed concentration (for an eight-hour averaging time). Current PM_{10} concentrations are at 52 percent of the allowed ambient standard. Nitrogen dioxide concentrations are at 36 percent of the allowed values. All other pollutants, for which ambient air quality standards exist, are below 25 percent of the allowed values.

4.7.3 Radiological Air Quality

Radionuclide emissions to the atmosphere from the Hanford Site have been steadily decreasing over the last few years as site operations have changed emphasis from the historical mission of materials production and processing to energy and waste management research. During 1992, all operations at the Hanford Site released less than 100 Ci of radionuclides to the atmosphere, most of which consisted of tritium and noble gases (Woodruff and Hanf 1993). Of that total, fission and activation products accounted for less than 0.036 Ci, uranium isotopes accounted for less than 1×10^{-6} Ci, and transuranics contributed less than 0.005 Ci. These releases resulted in a dose to the maximally exposed offsite resident of less than 0.005 mrem, which is several orders of magnitude less than the current EPA standard of 10 mrem per year for DOE facilities.

Ambient air monitoring for radionuclides consisted of sampling at 42 onsite and offsite locations during 1992. Total concentrations of alpha- and beta-emitting radionuclides at the site perimeter were indistinguishable from those at distant locations that are unaffected by Hanford emissions. Concentrations of two specific radionuclides (tritium and iodine-129) were elevated relative to background; however, their contribution to the total airborne activity was small.

4.8 Water Resources

4.8.1 Surface Water

4.8.1.1 Surface Water Hydrology. The Pasco Basin occupies about 4900 square kilometers (1900 square miles) and is located centrally within the Columbia Basin. Elevations within the Pasco Basin are generally lower than other parts of the plateau, and surface drainage enters it from other basins. Within the Pasco Basin, the Columbia River is joined by three major tributaries: the Yakima River, the Snake River, and the Walla Walla River.

The Hanford Site occupies approximately one-third of the land area within the Pasco Basin. Primary surface-water features associated with the Hanford Site are the Columbia and Yakima rivers. Several surface ponds and ditches are present, and they are generally associated with fuel- and waste-processing activities. Several small spring-streams occur on the Arid Land Ecology site on the western side of the Hanford Site.

A network of dams and multipurpose water resources projects is located along the course of the Columbia River. The principal dams are shown in Figure 4-13. Storage behind Grand Coulee Dam, combined with storage upstream in Canada, totals 3.1 x 10^{10} cubic meters (1.1 x 10^{12} cubic feet) of usable storage to regulate the Columbia River for power, flood control, and irrigation of land within the Columbia Basin project.

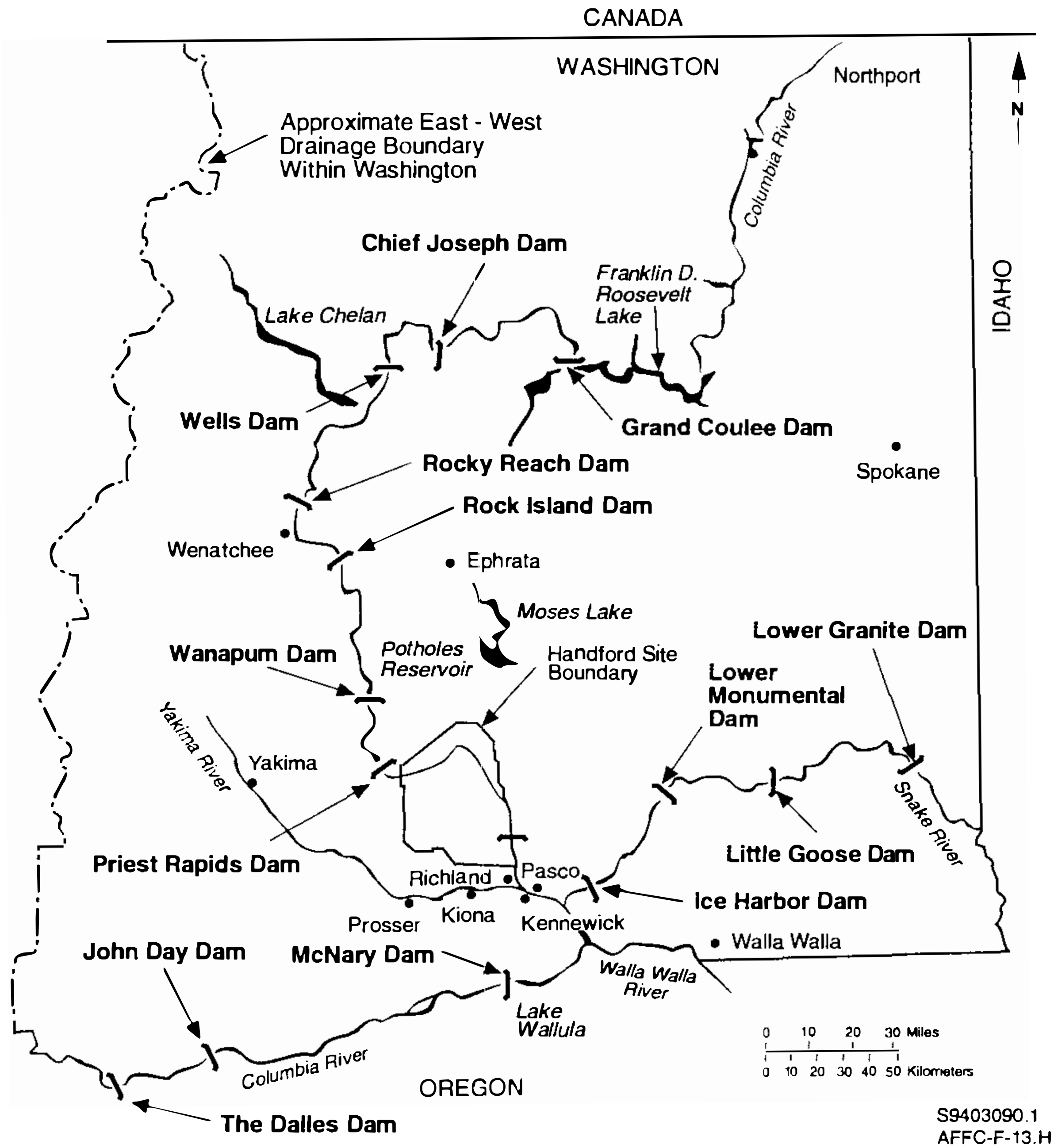

Figure 4-13. Locations of major surface water resources and principal dams within the Columbia Plateau.

Approximately two-thirds of the surface runoff, if there were any from Hanford, would drain directly into the Columbia River along the Hanford Reach, which extends from the upstream end of Lake Wallula to the Priest Rapids Dam. One-third of the surface runoff would drain into the Yakima River, which flows into the Columbia River below the Hanford Site. The flow has been inventoried and described in detail by the U.S. Army Corps of Engineers (DOE 1986a). Flow along this reach is controlled by the Priest Rapids Dam. Several drains and intakes are also present along this reach. These include irrigation outfalls from the Columbia Basin Irrigation Project and Hanford Site intakes for the onsite water export system.

Recorded flow rates of the Columbia River have ranged from 4500 to 18,000 cubic meters per second (~158,900 to 635,600 cubic feet per second) during the runoff in spring and early summer, to 1000 to 4500 cubic meters per second (35,300 to 158,900 cubic feet per second) during the low flow period of late summer and winter. The average annual Columbia River flow in the Hanford Reach, based on records from 65 years, is about 3400 cubic meters per second (120,100 cubic feet per second) (DOE 1988). A minimum flow of about 1020 cubic meters per second (35,000 cubic feet per second) is maintained along the Hanford Site. Normal river elevations within the site range from 120 meters (394 feet) above mean sea level where the river enters the Hanford Site near Vernita to 104 meters (341 feet) where it leaves the site near the 300-Area.

The Yakima River, near the southern portion of the Hanford Site, has a low annual flow compared to the Columbia River. For 57 years of record, the average annual flow of the Yakima River is about 104 cubic meters per second (3673 cubic feet per second) with monthly maximum and minimum flows of 490 cubic meters per second (17,305 cubic feet per second) and 4.6 cubic meters per second (162 cubic feet per second), respectively.

Cold Creek and its tributary, Dry Creek, are ephemeral streams within the Yakima River drainage system along the southern boundary of the Hanford Site. Both streams drain areas to the west of the Hanford Site and cross the southwestern part of the site toward the Yakima River.

Surface flow, when it occurs, infiltrates and disappears into the surface sediments in the western part of the Hanford Site (refer to subsection 4.6.1.3 for a discussion of soil types and moisture percolation). Rattlesnake Springs, located on the western part of the site, forms a small surface stream that flows for about 3 kilometers (1.8 miles) before disappearing into the ground. Approximately one-third of the Hanford Site is drained by the Yakima River system.

Total estimated precipitation over the Pasco Basin is about 9×10^6 cubic meters (318×10^6 cubic feet) annually, averaging less than 20 centimeters per year (~8 inches per year). Mean annual runoff from the basin is estimated to be less than 3.1×10^7 cubic meters per year (109×10^7 cubic feet per year), or approximately 3 percent of the total precipitation. The basin-wide runoff coefficient is zero for all practical purposes. The remaining precipitation is assumed to be lost through evapotranspiration, with a small component (perhaps less than 1 percent) recharging the groundwater system (DOE 1988).

Water use in the Pasco Basin is primarily from surface diversion with groundwater diversions accounting for less than 10 percent of the use. A listing of surface water diversions, volumes, types of usage, and the populations served is given in DOE (1988). Industrial and agricultural usage represent about 32 percent and 58 percent, respectively, and municipal use about 9 percent. The Hanford Site uses about 81 percent of the water withdrawn for industrial purposes. However, because of the N Reactor shutdown and considering the data in DOE (1988), these percentages now approximate 13 percent for industrial, 75 percent for agricultural, and 12 percent for municipal use, with the Hanford Site accounting for about 41 percent of the water withdrawn for industrial use.

Approximately 50 percent of the wells in the Pasco Basin are for domestic use and are generally shallow (less than 150 meters [500 feet]). Agricultural wells, used for irrigation and stock supply, make up the second-largest category of well use, about 24 percent for the Pasco Basin. Industrial users account for only about 3 percent of the wells (DOE 1988).

Most of the water used by the Hanford Site is withdrawn from the Columbia River. The principal users of groundwater within the Hanford Site are the Fast Test Flux Facility, with a 1988 use of 142,000 cubic meters (5.0×10^6 cubic feet) from two wells in the unconfined aquifer, and the PNL Observatory, with a water supply from a spring on the side of Rattlesnake Mountain.

Regional effects of water-use activities are apparent in some areas where the local water tables or potentiometric levels have declined because of withdrawals from wells. In other areas, water levels in the shallow aquifers have risen because of artificial recharge mechanisms, such as excessive application of imported irrigation water or impoundment of streams. Wastewater ponds on the Hanford Site have artificially recharged the unconfined aquifer below the 200-East and 200-West Areas. The increase in water table elevations was most rapid from 1950 to 1960, and apparently had nearly reached equilibrium between the unconfined aquifer and the recharge during 1970 to 1980 when only small increases in water table elevations occurred. Wastewater discharges from the 200-West Area were significantly reduced in 1984 (DOE 1988), with an accompanying decline in water table elevations.

4.8.1.2 Flood Plains. Large Columbia River floods have occurred in the past (DOE 1987), but the likelihood of recurrence of large-scale flooding has been reduced by the construction of several flood control/water storage dams upstream of the site. Major floods on the Columbia River are typically the result of rapid melting of the winter snowpack over a wide area augmented by above-normal precipitation. The maximum historical flood on record occurred June 7, 1894, with a peak discharge at the Hanford Site of 21,000 cubic meters per second (742,000 cubic feet per second). The flood plain associated with the 1894 flood is shown in Figure 4-14. The largest recent flood took place in 1948 with an observed peak discharge of 20,000 cubic meters per second (706,280 cubic feet per second) at the Hanford Site. The probability of flooding at the magnitude of the 1894 and 1948 floods has been greatly reduced because of upstream regulation by dams.

The Federal Emergency Management Agency has not prepared flood plain maps for the Hanford Reach of the Columbia River because that agency prepares maps only for developing areas (a criteria that specifically excludes the Hanford Reach).

Evaluation of flood potential is conducted in part through the concept of the probable maximum flood, determined from the upper limit of precipitation falling on a drainage area and other hydrologic factors, such as antecedent moisture conditions, snowmelt, and tributary conditions, that could result in maximum runoff. The probable maximum flood for the Columbia River below Priest Rapids Dam has been calculated to be 40,000 cubic meters per second (1.4 million cubic feet per second) and is greater than the 500-year flood. The flood plain associated with the probable maximum flood is shown in Figure 4-15. This flood would inundate parts of the 100-Areas located adjacent to the Columbia River, but the central portion of the Hanford Site where the SNF facility would be located would remain unaffected (DOE 1986a).

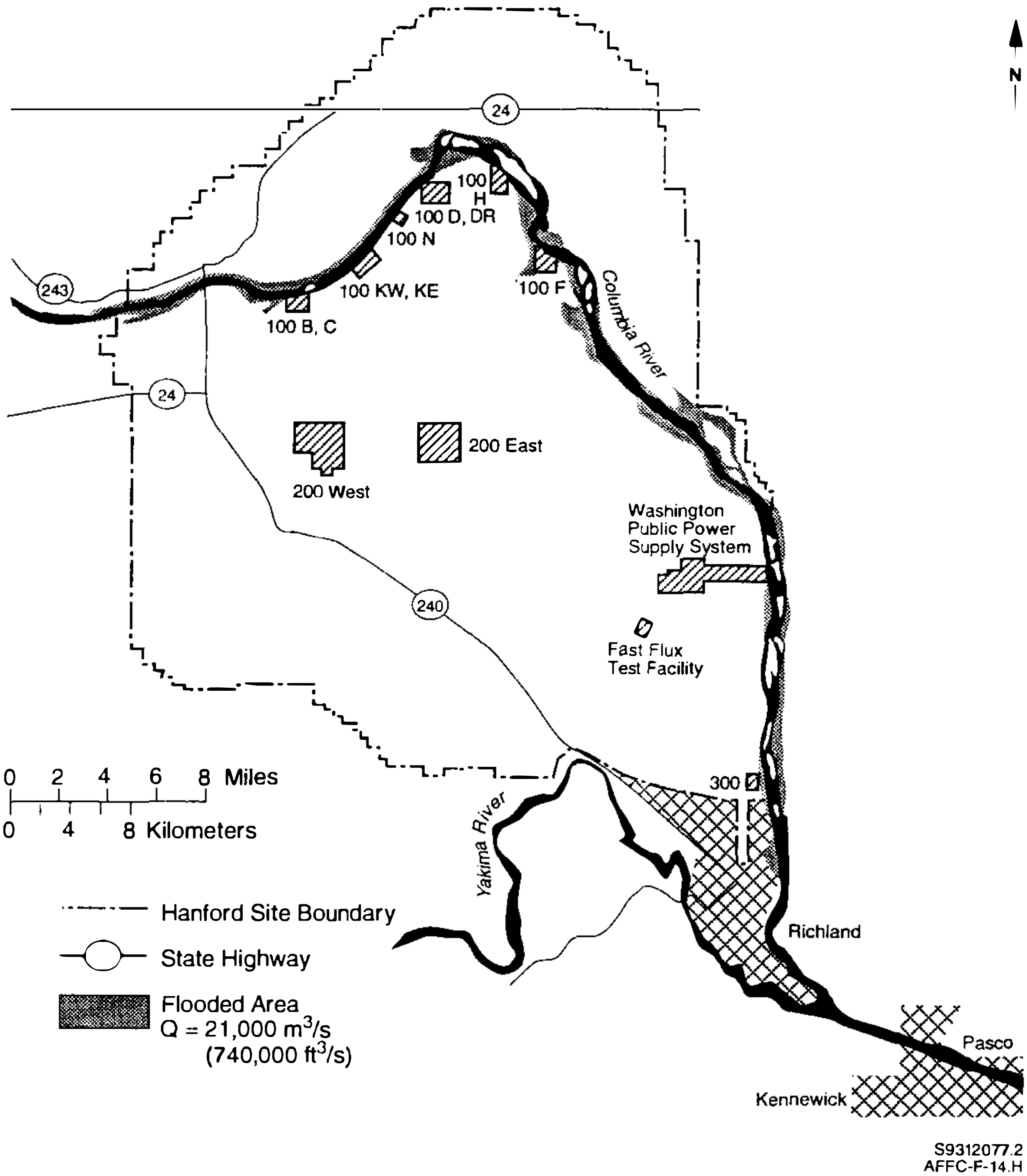

Figure 4-14. Flood area during the 1894 flood.

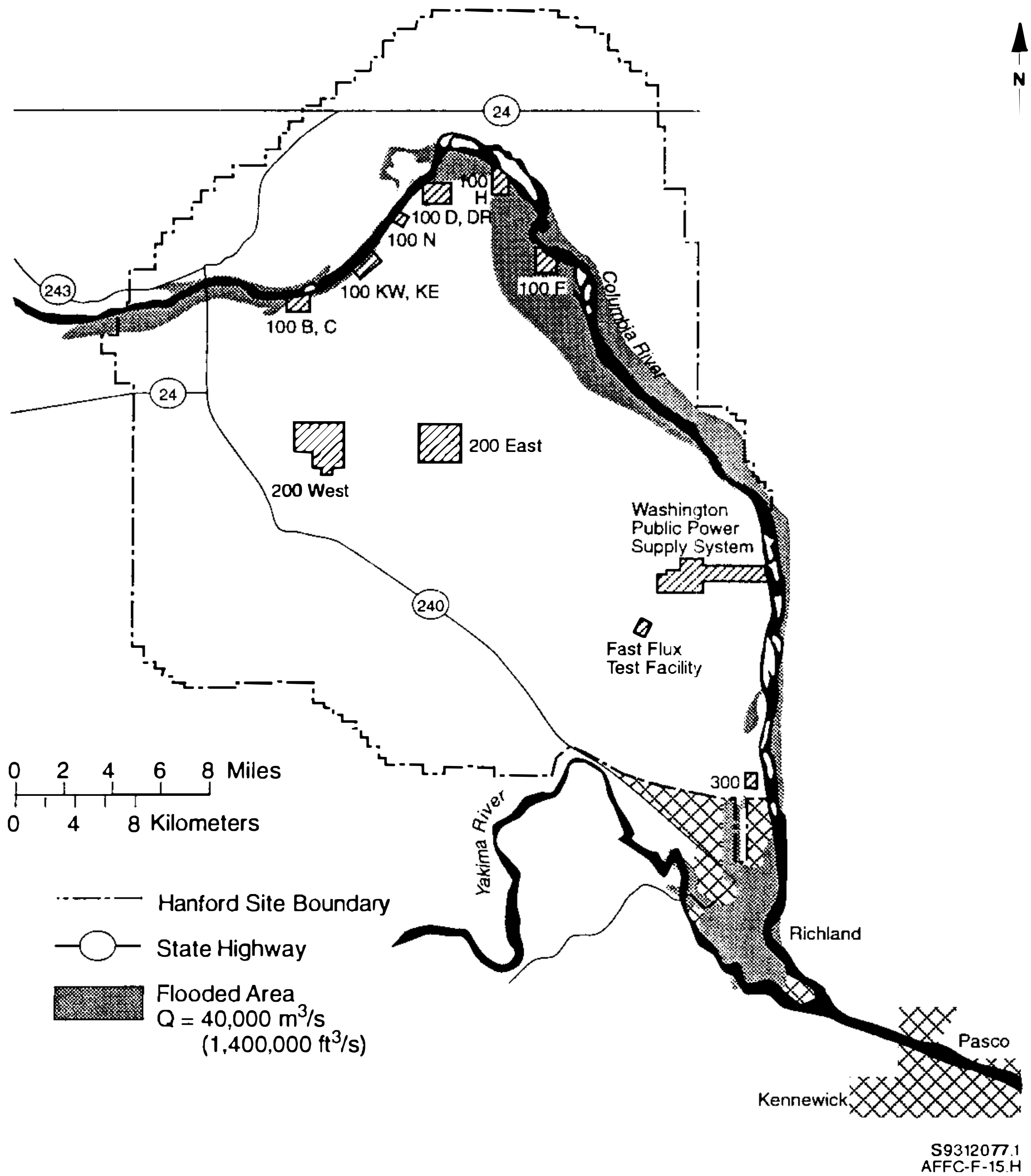

Figure 4-15. Flood area for the probable maximum flood.

The U.S. Army Corps of Engineers (1989) has derived the Standard Project Flood with both regulated and unregulated peak discharges given for the Columbia River below Priest Rapids Dam. Frequency curves for both natural (unregulated) and regulated peak discharges are also given for the same portion of the Columbia River. The regulated Standard Project Flood for this part of the river is given as 15,200 cubic meters per second (54,000 cubic feet per second) and the 100-year regulated flood as 12,400 cubic meters per second (440,000 cubic feet per second). No maps for the flooded areas are provided.

Potential dam failures on the Columbia River have been evaluated (DOE 1986a; ERDA 1976). Upstream failures could arise from a number of causes, with the magnitude of the resulting flood depending on the degree of breaching at the dam. The U.S. Army Corps of Engineers evaluated a number of scenarios on the effects of failures of Grand Coulee Dam, assuming flow conditions of the order of 11,000 cubic meters per second (400,000 cubic feet per second). For purposes of emergency planning, they hypothesized that 25 percent and 50 percent breaches, the instantaneous disappearance of 25 percent or 50 percent of the center section of the dam, would result from the detonation of nuclear explosives in sabotage or war. The discharge or floodwave resulting from such an instantaneous 50 percent breach at the outfall of the Grand Coulee Dam was determined to be 600,000 cubic meters per second (21 million cubic feet per second). In addition to the areas inundated by the probable maximum flood (see Figure 4-15), the remainder of the 100 Areas, the 300 Area, and nearly all of Richland, Washington, would be flooded (DOE 1986a; ERDA 1976). Determinations were not made for failures of dams upstream, for associated failures downstream of Grand Coulee, or for breaches greater than 50 percent of Grand Coulee for two principal reasons: the 50 percent scenario was believed to represent the largest realistically conceivable flow resulting from either a natural or human-induced breach (DOE 1986a); that is, it was hard to imagine that a structure as large as the Grand Coulee Dam would be 100 percent destroyed instantaneously. It was also assumed that such a scenario as the 50 percent breach would only occur as the result of direct explosive detonation, not because of a natural event such as an earthquake. Even a 50 percent breach under these conditions would indicate an emergency situation where other overriding major concerns might be present.

The possibility of a landslide resulting in river blockage and flooding along the Columbia River has also been examined for an area bordering the east side of the river upstream from the city of Richland (DOE 1986a). The possible landslide area considered was the 75-meter- (250-foot-) high bluff generally known as White Bluffs. Calculations were made for an 8×10^5 cubic meter (1×10^6 cubic yards) landslide volume with a concurrent flood flow of 17,000 cubic meters per second (600,000 cubic feet per second) (a 200-year flood) resulting in a flood wave crest elevation of 122 meter (400 foot) above mean sea level. Areas inundated upstream from such a landslide event would be similar to those shown in Figure 4-15.

A flood risk analysis of Cold Creek was conducted in 1980 as part of the characterization of a basaltic geologic repository for high-level radioactive waste. Such design work is usually done to the criteria Standard Project Flood or Probable Maximum Flood rather than the worst case or 100-year flood scenario. Therefore, in lieu of 100- and 500-year floodplain studies, a probable maximum flood evaluation was made for a reference repository location directly west of the 200-East Area and encompassing the 200-West Area (Skaggs and Walters 1981). Figure 4-16 shows the extent of this evaluation.

4.8.1.3 Surface Water Quality.

4.8.1.3.1 Water Quality of the Columbia River—The Department of Ecology classifies the Columbia River as Class A (excellent) between Grand Coulee Dam and the mouth of the river near Astoria, Oregon (DOE 1986a). The Hanford Reach of the Columbia River is the last free-flowing portion of the river in the United States.

Pacific Northwest Laboratory conducts routine monitoring of the Columbia River for both radiological and nonradiological water quality parameters. A yearly summary of results has been published since 1973 (Woodruff and Hanf 1993). Numerous other water quality studies have been conducted on the Columbia River relative to the impact of the Hanford Site during the past 37 years. Currently, eight outfalls are covered by National Pollutant Discharge Elimination System (NPDES) permits at the Hanford Site: two at the 100-K Area, five at the 100-N Area, and one at the 300 Area. These discharge locations are monitored for various measures of water quality, including nonradioactive and radioactive pollutants. The dose from any radionuclide releases is estimated for the Annual Environmental Monitoring Report for the Hanford Site. In 1993, monitored liquid discharges resulted in a dose of 0.012 mrem to the downstream maximally exposed individuals (Dirkes et al. 1994). Permit applications have been

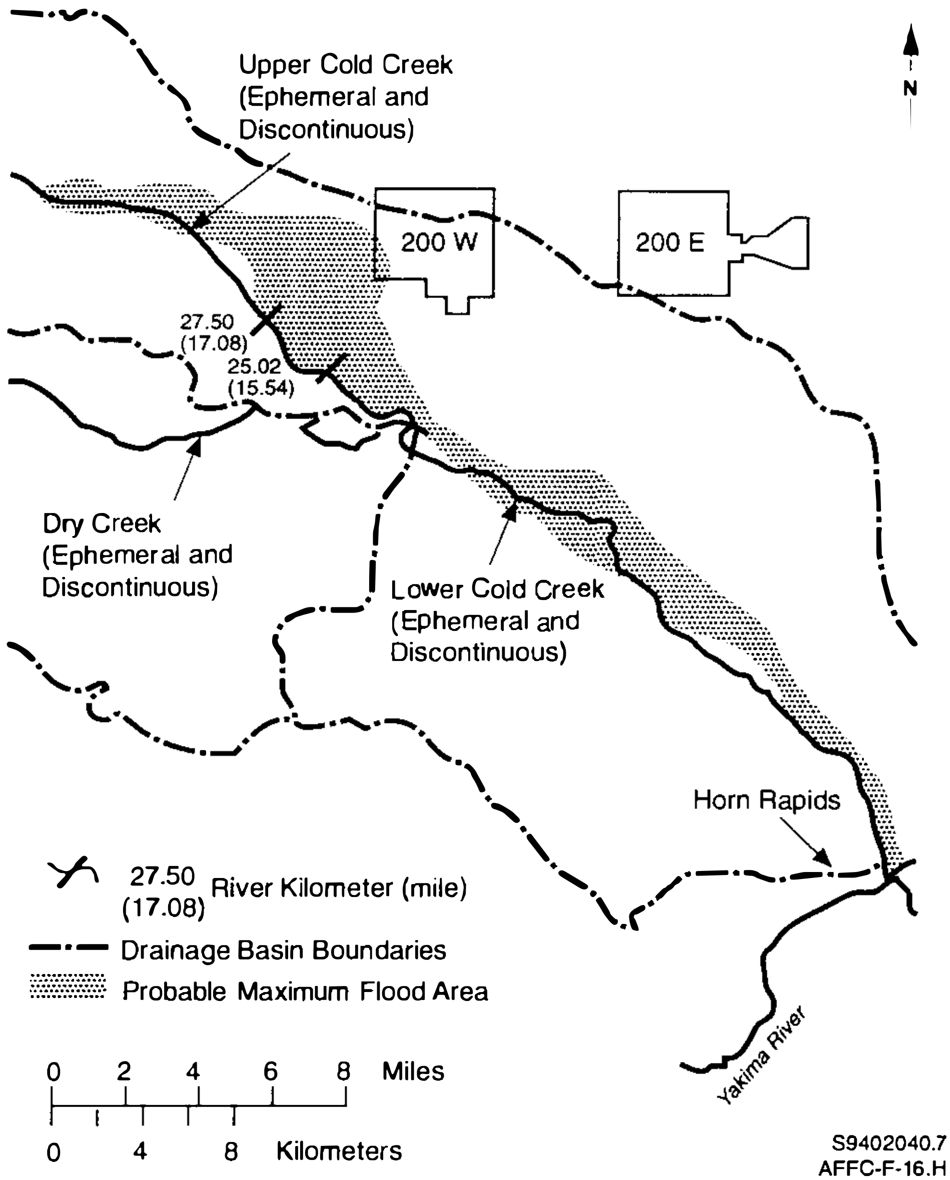

Figure 4-16. Extent of probable maximum flood in Cold Creek area.

submitted to EPA Region 10 for three new facilities (outfalls) planned for the 100 and 300 Areas. These new facilities include a treatment facility for process wastewater (1325-N), a filter backwash/ash sluicing wastewater disposal facility (315/384), and the 300 Area Treated Effluent Disposal Facility.

Radiological monitoring shows low levels of radionuclides in samples of Columbia River water. Tritium, iodine-129, and uranium are found in somewhat higher concentrations downstream of the Hanford Site than upstream (Woodruff and Hanf 1993), but well below concentration guidelines established by DOE and EPA drinking-water standards (Table 4.8-1). Cobalt-60 and iodine-131 were not consistently found in measurable quantities during 1989 in samples of Columbia River water from Priest Rapids Dam, the 300-Area water intake, or the Richland city pumphouse (Woodruff and Hanf 1991). In 1989, the average annual strontium-90 concentrations were essentially the same at Priest Rapids Dam (upstream of the Hanford Site) and the Richland Pumphouse (Woodruff and Hanf 1991).

Nonradiological water quality parameters measured during 1989 were similar to those reported in previous years and were within Washington State Water Quality Standards (Woodruff and Hanf 1991). Under Federal Water Pollution Control Act Amendments of 1972 (as amended by the Clean Water Act of 1972) the NPDES can regulate permits issued to DOE-RL for discharges of nonradioactive effluents made to the Columbia River.

Table 4.8-1. Annual average concentrations of radionuclides in Columbia River water during 1992.[a]

	Water concentrations (pCi/L)		
Radionuclides	Upstream concentration (Priest Rapids Dam)	Downstream concentration (Richland Pumphouse)	EDA drinking water standard
H-3	50	101	20,000
Sr-90	0.09	0.09	8.0
Uranium	0.42	0.51	NA
Tc-99	0.10	0.21	900
I-129	$<2.3 \times 10^{-5}$	$<1.4 \times 10^{-4}$	1

a. Data taken from Woodruff and Hanf (1993).

4.8.1.3.2 Water Quality of the Unconfined Aquifer—As part of the continuing environmental monitoring program, groundwater monitoring reports have been issued since 1956 and are now published in the Hanford Site Environmental Report, which is issued by calendar year. The shallow, unconfined aquifer in the Pasco Basin and on the Hanford Site contains waters of a dilute (less than or approximately 350 milligrams per liter total dissolved solids) calcium bicarbonate chemical type. Other principal constituents include sulfate, silica, magnesium, and nitrate. Variability in chemical composition exists within the unconfined aquifer in part because of natural variation in the composition of the aquifer material; in part because of agricultural and irrigation practices north, east, and west of the Hanford Site; and, on the Hanford Site, in part because of liquid waste disposal.

Graham et al. (1981) compared analyses of unconfined aquifer water samples taken by the U.S. Geological Survey in the Pasco Basin, but off the Hanford Site, with samples taken by PNL and the USGS on the Hanford Site for the years 1974 through 1979. In general, Hanford Site groundwater analyses showed higher levels of chemical constituents and temperatures than were reflected in the analyses of offsite samples.

Elevated levels of some constituents in the Hanford groundwater result from releases of various liquid wastes from disposal facilities, primarily in the 100 Areas (formerly the site of production reactor operations) and 200 Areas (formerly the spent fuel reprocessing and defense materials production site). Mobile contaminants, such as tritium and nitrate, from the 200 Areas are present in a groundwater plume that extends across the southeastern quadrant of the Hanford Site and enters the Columbia River along a broad front north of the 300 Area. Contaminants having lower mobility are generally confined to smaller localized plumes in the vicinity of the disposal facilities and migrate more slowly toward the Columbia River (Dirkes et al. 1994). Some longer-lived radionuclides, such as strontium-90 and cesium-137, have reached the groundwater, primarily through liquid waste disposal cribs. Minor quantities of longer-lived radionuclides have also reached the water table via a failed groundwater monitoring well casing and through reverse well injection, a disposal practice that was discontinued at Hanford in 1947 (Smith 1980).

Of the contaminants found in groundwater, several radionuclides and nonradioactive chemicals were present in concentrations that exceeded EPA drinking water standards or DOE Derived Concentration Guides (DCG) in 1993 (Dirkes et al. 1994). These quantities are used as a relative measure of contamination, although with one exception, groundwater beneath the site is not used for human consumption or food production. Groundwater utilized for drinking at the FFTF visitor center contains above-background quantities of tritium and iodine-129 from the 200 Area plume; however, these levels are well below the EPA drinking water standards. There is little opportunity for contaminated groundwater to migrate to locations where members of the public might utilize it directly for domestic purposes or irrigation. Groundwater in the unconfined aquifer beneath the Hanford Site is relatively isolated, and generally flows toward the north and east where it discharges to the Columbia River. Normal hydraulic gradients within the unconfined aquifer beneath the Hanford Site prevent southward migration of groundwater toward populated areas near Richland, and recharge to the Columbia River from aquifers in Franklin County to the north and east prevents radionuclides in the Columbia River from migrating to groundwater across the river from Hanford.

Groundwater monitoring at the 100 Areas detected concentrations of cobalt-60, strontium-90, antimony-125, and uranium that were above the EPA drinking water standards. Tritium concentrations exceeded both the EPA drinking water standard and the DOE DCG at one sample well in each of the 100-N and 100-K Areas. In 200 Area wells, cobalt-60, technetium-99, iodine-129, cesium-137, uranium, and plutonium were occasionally found in concentrations that exceeded the EPA drinking water standard; tritium and strontium-90 exceeded both the EPA drinking water standard and the DOE DCG in some locations. Only uranium exceeded the EPA drinking water standard in 300 Area wells, a result of liquid waste disposal at former fuel fabrication facilities.

Three nonradiological constituents - nitrate, chromium, and trichloroethylene - exceeded EPA drinking water standards in both 100 and 200 Area groundwater. In addition to those constituents, some 200 Area wells exceeded EPA drinking water standards for cyanide, fluoride, carbon tetrachloride, and chloroform. Only trichloroethylene was found above the drinking water limits in the 300 Area.

The occurrence and consequences of leaks from waste storage tanks and of radioactive materials in soils have been described elsewhere (ERDA 1975). These occurrences have not resulted, and are not expected to result, in radiation exposure to the public (ERDA 1975; DOE 1987). Leakage from the 105-KE fuel storage basin results in groundwater contamination with several radionuclides, as noted previously. The more mobile radionuclides reach the Columbia River via springs near the 100-K Area, although radionuclides in the springs were below the EPA drinking water standard in 1993 (Dirkes et al. 1994).

Radioactive and nonradioactive effluents are discharged to the environment from Westinghouse Hanford Company facilities in the 200 Area (Cooney et al. 1988). These effluents, in general, are discharged to the soil column. Cooling water represents by far the largest volume of potentially radioactive liquid effluent. Additional treatment systems for these effluents are being designed and installed pursuant to the schedule set forth in the Hanford Federal Facility Agreement and Consent Order, which was jointly issued by DOE, EPA, and the Washington Department of Ecology in May 1989. Under the provisions of the Comprehensive Environmental Response Compensation and Liability Act, remedial investigations/feasibility studies will be conducted for groundwater operable units at Hanford.

Springs are common on basalt ridges surrounding the Pasco Basin. Geochemically, spring waters are of a calcium or sodium bicarbonate type with low dissolved solids (approximately 200 to 400 milligrams per liter) (DOE 1986a). Compositionally these waters are similar to shallow local groundwaters (unconfined aquifer and upper Saddle Mountains basalt). However, they are readily distinguishable from waters of the lower Saddle Mountains (Mabton interbed) and the Wanapum and Grande Ronde basalts, which are of sodium bicarbonate to sodium chloride bicarbonate (or sodium chloride sulfate) type. Currently, no evidence suggests these spring waters contain any significant component of deeper groundwater.

4.8.1.3.3 Water Quality of the Confined Aquifer—Areal and stratigraphic changes in groundwater chemistry characterize basalt groundwaters beneath the Hanford Site (Graham et al. 1981). The stratigraphic position of these changes is believed to delineate flow-system boundaries and to identify chemical evolution taking place along groundwater flow paths. Using these data, some potential mixing of groundwaters has also been located; however, the rate of mixing is unknown. According to Woodruff and Hanf (1993), no evidence of contamination was observed in the groundwater of the confined aquifer on Rattlesnake Ridge. Groundwater in one well in this aquifer contained 8,800 micrograms of nitrate per liter in 1992. The well was located near an erosional window in the confining basalt flow. In another well, tritium levels were elevated (maximum of 7,830 picocuries per liter) in 1992. In the same well, elevated levels of iodine-129 (0.15 picocuries per liter) were observed in 1992.

4.8.2 Groundwater

4.8.2.1 Groundwater Hydrology. The regional geohydrologic setting of the Pasco Basin is based on the stratigraphic framework consisting of numerous Miocene tholeiitic flood basalts of the Columbia River Basalt group; relatively minor amounts of intercalated fluvial and volcanoclastic Ellensburg Formation sediments; and fluvial, lacustrine, and glaciofluvial suprabasalt sediments. The vertical order of the geological units from the surface downward is Hanford formation, Middle Ringold Formation, Lower Ringold Formation, Basal Ringold Formation, and bedrock, e.g., basalt. Figure 4-3 illustrates the stratigraphic layering of the hydrogeologic units underlying the Hanford Site, and Figure 4-17 shows the order of the geological units. The surface Hanford formation varies in thickness across the Hanford Site from approximately 15 to 100 meters (49 to 328 feet) thick (Figure 4-17). The Middle Ringold Formation varies from 10 to 110 meters (33 to 361 feet) thick. The Lower Ringold and Basal Ringold Formations extend eastward from the western boundary of the site approximately 1.1 kilometers (6.8 miles). The Lower Ringold Formation is rather uniform in thickness at 20 meters (66 feet), while the Basal Ringold Formation demonstrates a maximum thickness of 40 meters (131 feet) at the far western boundary of the site (interpolated from Woodruff and Hanf 1993). Lateral groundwater movement is known to occur within a shallow, unconfined

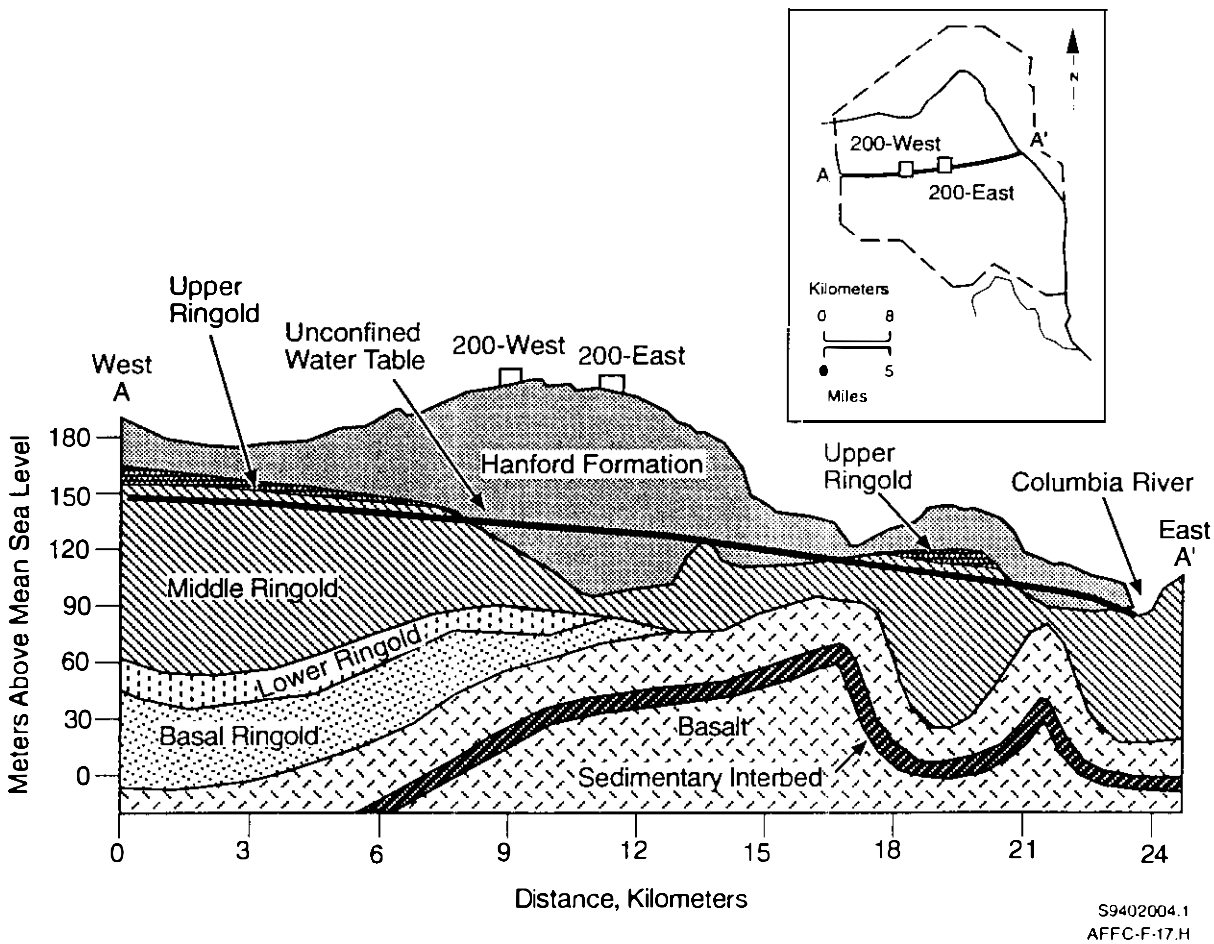

Figure 4-17. Geologic cross section of the Hanford Site (modified from Tallman et al. 1979).

aquifer consisting of fluvial and lacustrine sediments lying on top of the basalts, and within deeper confined-to-semiconfined aquifers consisting of basalt flow tops, flow bottom zones, and sedimentary interbeds (DOE 1988). These deeper aquifers are intercalated with aquitards consisting of basalt flow interiors. Vertical flow and leakage between geohydrologic units is inferred and estimated from water level or potentiometric surface data but is not quantified, and direct measurements are not available (DOE 1988).

The multiaquifer system within the Pasco Basin has been conceptualized as consisting of four geohydrologic units: (1) the Grande Ronde Basalt; (2) Wanapum Basalt; (3) Saddle Mountain Basalt; and (4) suprabasalt Hanford and Ringold Formation sediments. Geohydrologic units older than the Grande Ronde Basalt are probably of minor importance to the regional hydrologic dynamics and system.

The Grande Ronde Basalt is the most voluminous and widely spread formation within the Columbia River Basalt group and has a thickness of at least 2745 meters (9000 feet). The Grande Ronde Basalt geohydrologic unit is composed of the Grande Ronde Basalt and minor intercalated sediments equivalent to or part of the Ellensburg Formation (DOE 1988). More than 50 flows of Grande Ronde Basalt underlie the Pasco Basin, but little is known of the lower 2200 to 2500 meters of this geohydrologic unit. This unit is a confined-to-semiconfined flow system that is recharged along the margins of the Columbia Plateau where the unit is at or close to the land surface, and by surface-water and groundwater inflow from lands adjoining the plateau. Vertical movement into and out of the unit is known to occur. Groundwater within the unit in the eastern Pasco Basin is believed to be derived from groundwater inflow from the east and northeast.

The Wanapum Basalt geohydrologic unit consists of basalt flows of the Wanapum Basalt intercalated with minor and discontinuous sedimentary interbeds of the Ellensburg Formation or equivalent sediments. In the Pasco Basin, the Wanapum Basalt consists of three members, each consisting of multiple flows. The geohydrologic unit underlies the entire Pasco Basin and has a maximum thickness of 370 meters (1215 feet). Groundwater within the Wanapum Basalt geohydrologic unit is confined to semiconfined. Recharge is believed to occur from precipitation where the Wanapum Basalt is not overlain by great thicknesses of younger basalt, leakage from adjoining formations, and surface-water and groundwater inflow from lands adjoining the plateau. Local recharge is derived from irrigation. Within the Pasco Basin, recharge occurs along the anticlinal ridges to the north and west, with recharge in the eastern basin being from groundwater inflow from the east and northeast (DOE 1988). Interbasin transfer and vertical leakage are also believed to contribute to the recharge.

The Saddle Mountains Basalt geohydrologic unit is composed of the youngest formation of the Columbia River Basalt Group and several thick sedimentary beds of the Ellensburg Formation or equivalent sediments that comprise up to 25 percent of the unit. Within the Pasco Basin, the Saddle Mountains Basalt contains seven members, each with one or more flows. This geohydrologic unit underlies most of the Pasco Basin, attaining a thickness of about 290 meters (950 feet), but is absent along the northwest part of the basin and along some anticlinal ridges. Groundwater in the Saddle Mountains geohydrologic unit is confined to semiconfined, with recharge and discharge believed to be local (DOE 1988).

The rock materials that overlie the basalts in the structural and topographic basins within the Columbia Plateau generally consist of Miocene-Pliocene sediments, volcanics, Pleistocene sediments (including those from catastrophic flooding), and Holocene sediments consisting mainly of alluvium and eolian deposits. The suprabasalt geohydrologic unit (referred to as the Hanford/Ringold unit) consists principally of the Miocene-Pliocene Ringold Formation stream, lake, and alluvial materials, and the Pleistocene catastrophic flood deposits informally called the Hanford formation. Groundwater within the suprabasalt geohydrologic unit is generally unconfined, with recharge and discharge usually coincident with topographic highs and lows (DOE 1988). The Hanford/Ringold unit is essentially restricted to the Pasco Basin with principal recharge occurring along the periphery of the basin from precipitation and ephemeral streams.

Little if any natural recharge occurs within the Hanford Site, but artificial recharge occurs from liquid waste disposal activities (Woodruff and Hanf 1993). Recharge from irrigation occurs east and north of the Columbia River and in the synclinal valleys west of the Hanford Site. Upward leakage from lower aquifers into the unconfined aquifer is believed to occur in the northern and eastern sections of the Hanford Site. Groundwater discharge is primarily to the Columbia River.

Groundwater under the Hanford Site occurs under unconfined and confined conditions (Figure 4-17). The unconfined aquifer is contained within the glaciofluvial sands and gravels of the Hanford formation and within the Ringold Formation. It is dominated by the middle member of the Ringold Formation, consisting of sands and gravels with varying amounts of cementation. The bottom of the unconfined aquifer is the basalt surface or, in some areas, the clay zones of the Lower Ringold. A semiconfined aquifer occurs in areas where the coarse-grained Basal Ringold lies between the basalt and the fine-grained Lower Ringold. The confined aquifers consist of sedimentary interbeds and/or interflow zones that occur between dense basalt flows in the Columbia River Basalt Group. The main water-bearing portions of the interflow zones occur within a network of interconnecting vesicles and fractures of the flow tops or flow bottoms.

4.8.2.2 Vadose Zone Hydrology. Sources of natural recharge to the unconfined aquifer are rainfall and runoff from the higher bordering elevations, water infiltrating from small ephemeral streams, and river water along influent reaches of the Yakima and Columbia rivers. In order to define the movement of water in the vadose zone, the movement of precipitation through the unsaturated (vadose) zone has been studied at several locations on the Hanford Site. Conclusions from these studies are varied depending on the location studied. Some investigators conclude that no downward percolation of precipitation occurs on the 200-Area Plateau where soil texture is varied and is layered with depth, and that all moisture penetrating the soil is removed by evaporation. Others have observed downward water movement below the root zone in tests conducted near the 300 Area, where soils are coarse textured and precipitation was above normal (DOE 1987).

From the recharge areas to the west, the groundwater flows downgradient to the discharge areas, primarily along the Columbia River. This general west-to-east flow pattern is interrupted locally by the groundwater mounds in the 200 Areas. From the 200 Areas, a component of groundwater also flows to the north, between Gable Mountain and Gable Butte. These flow directions represent current conditions; the aquifer is dynamic, and responds to changes in natural and artificial recharge.

Local recharge to the shallow basalts is believed to result from infiltration of precipitation and runoff along the margins of the Pasco Basin. Regional recharge of the deep basalts is thought to result from interbasin groundwater movement originating northeast and northwest of the Pasco Basin in areas where the Wanapum and Grande Ronde Basalts crop out extensively (DOE 1986a). Groundwater discharge from the shallow basalt is probably to the overlying unconfined aquifer and the Columbia River. The discharge area(s) for the deep groundwaters is presently uncertain, but flow is believed to be generally southeastward with discharge speculated to be south of the Hanford Site (DOE 1986a).

4.8.3 Existing Radiological Conditions

This section relates to the hydrology of the Hanford Site in general and to the hydrology of the 200 Area specifically because it is the location of the proposed SNF facility.

4.8.3.1 Hydrology of the Hanford Site. Groundwater quality on the Hanford Site has been affected by defense-related activities to produce nuclear materials. Due to the arid nature of the climate, natural recharge of the groundwater on the site is normally low. Artificial recharge has occurred in the past from the disposal of liquid waste associated with processing operations in the 100, 200, and 300 Areas that created mounds of water underlying discharge points. While most of the site does not have contaminated groundwater, large areas underlying the site do have elevated levels of both radiological and nonradiological constituents. The liquid effluents discharged into the ground have carried with them certain radionuclides and chemicals that move through the soil column at varying rates, eventually enter the groundwater, and form plumes of contamination (see Figure 5.54 in DOE 1992a).

Groundwater monitoring is conducted on an annual basis on the Hanford Site as part of the Hanford Ground-Water Environmental Surveillance Program and other monitoring programs to study the movement of plumes, groundwater quality, and the concentration of certain constituents as regulated by the EPA, the DOE, and Washington State. In 1992, several groundwater samples were taken from approximately 720 wells, of which 50 percent were sampled at least quarterly or more frequently. The remainder were sampled either once or twice. Figure 5.49 in DOE (1992a) illustrates the locations of these monitoring wells.

Results indicate that total alpha, total beta, tritium, cobalt-60, strontium-90, technetium-99, iodine-129, cesium-137, and uranium concentrations in wells in or near operating areas exceeded Drinking Water Standards (DWS) (see Tables C2 and C3 in Appendix C of DOE [1992a]). Concentrations of uranium in the 200-West Area, tritium in the general 200 Area, strontium-90 in the 100-N and 200-East Areas exceeded the Derived Concentration Guides (DCGs) [see Table C6 in Appendix C of DOE (1992b)]. Tritium continues to slowly migrate downgradient with the groundwater flow where it enters the Columbia River; 1 curie of tritium was discharged to the Columbia River from the 100 Areas in 1992 (Woodruff and Hanf 1993).

Nitrate concentrations also exceeded DWS at various locations in the 100, 200, and 300 Areas and at several 600 Area locations. Elevated concentrations were also detected for chromium, cyanide, carbon tetrachloride, chloroform, and trichloroethylene in various sample wells in the 100 and 200 Areas. For further information regarding groundwater quality on the Hanford Site, refer to DOE (1992b).

4.8.3.2 Hydrology of the 200 Areas. The unconfined aquifer beneath the Hanford Site is contained within the Ringold Formation and the overlying Hanford formation. The unconfined aquifer is affected by wastewater disposed to surface and subsurface disposal sites. The depth to groundwater ranges from 55 to 95 meters (180 to 310 feet) on the 200 Area Plateau. The bottom of the unconfined aquifer is the uppermost basalt surface or, in some areas, the clays of the Lower Ringold Member. The thickness of the unconfined aquifer in the 200 Areas ranges from less than 15 to 61 meters (50 to 200 feet). Beneath the unconfined aquifer is a confined aquifer system consisting of sedimentary interbeds or interflow zones that occur between dense basalt flows or flow units.

The sources of natural recharge to the unconfined aquifer are rainfall from areas of high relief to the west of the Hanford Site and two ephemeral streams, Cold Creek and Dry Creek. From the areas of recharge, the groundwater flows downgradient and discharges into the Columbia River. This general flow pattern is modified by basalt outcrops and subcrops in the 200 Areas and by artificial recharge.

The unconfined aquifer beneath the 200 Areas receives artificial recharge from liquid disposal areas. Cooling water disposed to ponds has formed groundwater mounds beneath two former and one continuing high-volume disposal sites: U Pond in the 200-West Area, B Pond east of the 200-East Area, and Gable Mountain Pond north of the 200-East Area. The water table rose approximately 20 meters (65 feet) under U Pond and 9 meters (30 feet) under B Pond compared with pre-Hanford conditions (Newcomb et al. 1972). However, U Pond and Gable Mountain Pond have been eliminated and, with no further recharge from them, the water levels will decline over the coming years. U Pond was deactivated in 1984 and Gable Mountain Pond was decommissioned and backfilled in 1987. The volume of B Pond increased after the elimination of Gable Mountain Pond.

The dry nature (for example, climate, waste form, and depth to water) of the low-level burial ground and the limited natural surface recharge available from precipitation minimize the probability of leachate formation and migration from these facilities.

Additional characterization and enhanced groundwater monitoring of the 200 Areas are currently being conducted pursuant to requirements established under the Resources Conservation and Recovery Act. When complete, this work will supply additional information on the 200 Areas.

4.8.4 Water Rights

The Hanford Site, situated along the Columbia River and near the Yakima River, lies within a region traditionally concerned about water rights. Typical water uses in this region include cooling a commercial nuclear power plant, irrigation, and municipal and industrial uses. Cooling water was withdrawn from the Columbia River to cool the defense reactors at Hanford. The DOE continues to assert a federally reserved water withdrawal right with respect to its existing Hanford operations. Current activities use water withdrawn from the Columbia River under the Department's federally reserved water right.

4.9 Ecological Resources

The Hanford Site is a relatively large, undisturbed area (1450 square kilometers [~560 square miles]) of shrub-steppe that contains numerous plant and animal species adapted to the region's semiarid environment. The site consists of mostly undeveloped land with widely spaced clusters of industrial buildings located along the western shoreline of the Columbia River and at several locations in the interior of the site. The industrial buildings are interconnected by roads, railroads, and electrical transmission lines. The major facilities and activities occupy about 6 percent of the total available land area, and their impact on the surrounding ecosystems is minimal. Most of the Hanford Site has not experienced tillage or livestock grazing since the early 1940s. The Columbia River flows through the Hanford Site, and although the river flow is not directly impeded by artificial dams within the Hanford Site, the historical daily and seasonal water fluctuations have been changed by dams upstream and downstream of the site (Rickard and Watson 1985). The Columbia River and other water bodies on the Hanford Site provide habitat for aquatic organisms. The Columbia River is also accessible for public recreational use and commercial navigation.

Topography of the proposed SNF facility site is level to gently sloping to the northeast. Substrate on the subject area is primarily Burbank loamy sand intergraded with Rupert sand. The latter consists of broad, stabilized sand dunes. Several used and unused unpaved roads cross the project area (Figure 4-18) with resulting disturbance to the plant community. The subject area outside the disturbed area is primarily a mature stand of big sagebrush with an understory of cheatgrass, an alien weed species, and Sandberg's bluegrass (Figure 4-18); there are approximately 494 square kilometers (191 square miles) of this community on the Hanford site. Sagebrush-bitterbrush/cheatgrass comprises the second largest plant community. Cover of big sagebrush increases rapidly from 10-25 percent near Route 4 to 25-50 percent over the remainder of the site. Cover of cheatgrass and Sandberg's bluegrass is mostly uniform across the subject area at 25-50 percent and 10-20 percent, respectively.

4.9.1 Terrestrial Resources

4.9.1.1 Vegetation. The Hanford Site, located in southeastern Washington, has been botanically characterized as a shrub-steppe. Because of the site's aridity, the productivity of both plants and animals is relatively low compared with other natural communities. In the early 1800s, the dominant plant in the area was big sagebrush with an understory of perennial bunch-grasses, especially Sandberg's bluegrass and bluebunch wheatgrass. With the advent of settlement that brought livestock grazing and crop raising, the natural vegetation mosaic was opened to a persistent invasion by alien annuals, especially cheatgrass. Today cheatgrass is the dominant plant on fields that were cultivated 50 years ago. Cheatgrass is also well established on rangelands at elevations less than 244 meters (800 feet) (Rickard and Rogers 1983). Wild-fires in the area are common; the most recent extensive fire in 1984 significantly altered the shrub component of the vegetation. The dryland areas of the Hanford Site were treeless in the years before land settlement; however, for several decades before 1943, trees were planted and irrigated on most of the farms to provide windbreaks and shade. When the farms were abandoned in 1943, some of the trees died but others have persisted, presumably because their

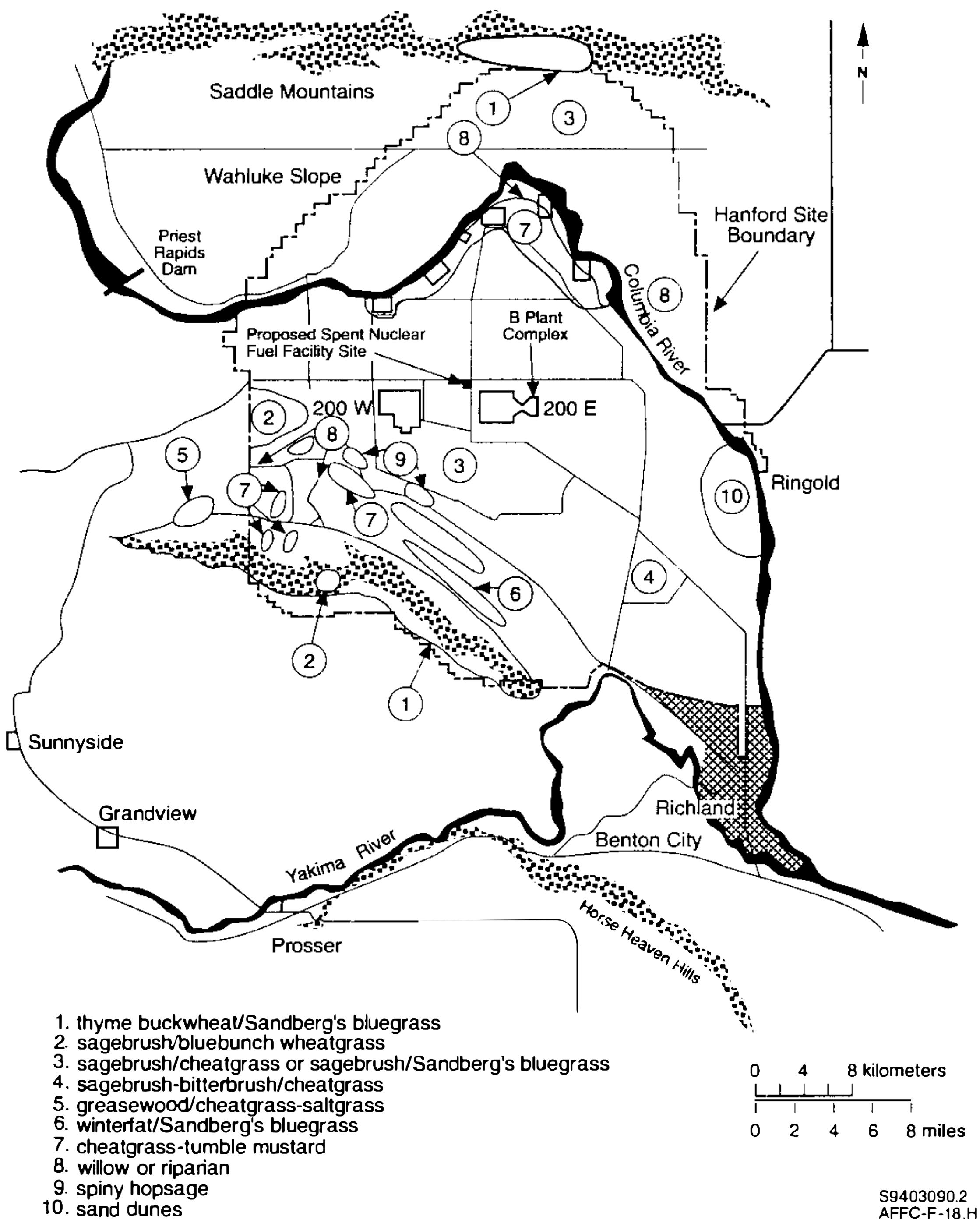

Figure 4-18. Distribution of vegetation types on the Hanford Site.

roots are deep enough to contact groundwater. Today these trees serve as nesting platforms for several species of birds, including hawks, owls, ravens, magpies, and great blue herons, and as night roosts for wintering bald eagles (Rickard and Watson 1985). The vegetation mosaic of the Hanford Site currently consists of 10 major kinds of plant communities:

1) thyme buckwheat/Sandberg's bluegrass
2) sagebrush/bluebunch wheatgrass
3) sagebrush/cheatgrass or sagebrush/Sandberg's bluegrass
4) sagebrush-bitterbrush/cheatgrass
5) greasewood/cheatgrass-saltgrass
6) winterfat/Sandberg's bluegrass
7) cheatgrass-tumble mustard
8) willow or riparian
9) spiny hopsage/Sandberg's bluegrass
10) sand dunes.

The dominant plant community on the proposed SNF site is sagebrush/Sandberg's bluegrass, with cheatgrass-tumble mustard occurring in the southern portion of the site. A table listing common plants on the Hanford Site can be found in Cushing (1992).

Almost 600 species of plants have been identified on the Hanford Site (Sackschewsky et al. 1992). The dominant plants on the 200 Area Plateau are big sagebrush, rabbitbrush, cheatgrass, and Sandberg's bluegrass, with cheatgrass providing half of the total plant cover. More than 100 species of plants have been identified in the 200 Area Plateau. Cheatgrass and Russian thistle, annuals introduced to the United States from Eurasia in the late 1800s, invade areas where the ground surface has been disturbed. Certain desert plants have roots that grow to depths approaching 10 meters (33 feet) (Napier 1982); however, root penetration to these depths has not been demonstrated for plants in the 200 Areas. Rabbitbrush roots have been found at a depth of 2.4 meters (8 feet) near the 200 Areas (Klepper et al. 1979). Mosses and lichens appear abundantly on the soil surface; lichens commonly grow on the shrub stems. The important desert shrubs, big sagebrush and bitterbrush, are widely spaced and usually provide less than 20 percent canopy cover. The important understory plants are grasses, especially cheatgrass, Sandberg's bluegrass, Indian ricegrass, June grass, and needle-and-thread grass.

As compared to other semiarid regions in North America, primary productivity is relatively low and the number of vascular plant species is also low. This situation is attributed to the low annual precipitation (16 centimeters [~6 inches]), the low water-holding capacity of the rooting substrate (sand), and the droughty summers and occasionally very cold winters.

Sagebrush and bitterbrush are easily killed by summer wildfires, but the grasses and other herbs are relatively resistant and usually recover in the first growing season after burning. Fire usually opens the community to wind erosion. The severity of erosion depends on the severity and areal extent of the fire. Hot fires incinerate entire shrubs and damage grass crowns. Less intensive fires leave dead stems standing, and recovery of herbs is prompt. The most recent and extensive wildfire occurred in the summer of 1984.

Bitterbrush shrubs provide browse for a resident herd of wild mule deer. Bitterbrush shrubs are slow to recolonize burned areas because invasion is by seeds. Bitterbrush does not sprout even when fire damage is relatively light.

Certain passerine birds (such as sage sparrow, sage thrasher, and loggerhead shrike) rely on sagebrush or bitterbrush for nesting. These birds are not expected to nest in places devoid of shrubs. Jackrabbits also appear to avoid burned areas without shrubs. Birds that nest on the ground in areas without shrubs included longbilled curlews, horned larks, Western meadowlarks, and burrowing owls.

An ecological inventory of the vegetation on the proposed SNF facility site revealed two primary vegetation types: burned and unburned sagebrush/cheatgrass. Two species predominated in the burned area: cheatgrass and tarweed fiddleneck; the unburned vegetation comprised mainly cheatgrass and big sagebrush. During the one-day survey, approximately 43 species were identified.

4.9.1.2 Insects. More than 300 species of terrestrial and aquatic insects have been found on the Hanford Site. Grasshoppers and darkling beetles are among the more conspicuous groups and, together with other species, are important in the food web of the local birds and mammals. Most species of darkling beetles occur throughout the spring to fall period, although some species are present only during two or three months in the fall (Rogers and Rickard 1977). Grasshoppers are evident during the late spring to fall. Both beetles and grasshoppers are subject to wide annual variations in abundance.

4.9.1.3 Reptiles and Amphibians. Among amphibians and reptiles, 12 species are known to occur on the Hanford Site (Fitzner and Gray 1991). The occurrence of these species is infrequent when compared with similar fauna of the southwestern United States. The side-blotched lizard is the most abundant reptile and can be found throughout the Hanford Site.

Short-horned and sagebrush lizards are also common in selected habitats. The most common snakes are the gopher snake, the yellow-bellied racer, and the Pacific rattlesnake, all found throughout the Hanford Site. Striped whipsnakes and desert night snakes are rarely found, but some sightings have been recorded for the site. Toads and frogs are found near the permanent water bodies and along the Columbia River. Cushing (1992) contains a list of all the reptiles and amphibians occurring on the Hanford Site.

4.9.1.4 Birds. Fitzner and Gray (1991) and Landeen et al. (1992) have presented data on birds observed on the Hanford Site. The horned lark and western meadowlark are the most abundant nesting birds in the shrub-steppe. A list of some of the more common birds present on the Hanford Site can be found in Cushing (1992).

4.9.1.4.1 Birds Inhabiting Terrestrial Habitats—The game birds inhabiting terrestrial habitats at Hanford are the chukar, gray partridge, and mourning dove. The chukar and partridge are year-round residents, but mourning doves are migrants. Although a few doves overwinter in southeastern Washington, most leave the area by the end of September. Mourning doves nest on the ground and in trees all across the Hanford Site. Chukars are most numerous in the Rattlesnake Hills, Yakima Ridge, Umtanum Ridge, Saddle Mountains, and Gable Mountain areas of the Hanford Site. A few birds also inhabit the 200-Area Plateau. Gray partridges are not as numerous as chukars, and their numbers also vary greatly from year to year. Sage grouse populations have declined on the Hanford Site since the 1940s, and it is probable there are no grouse nests on the site at this time. The nearest viable population is located on the U.S. Army's Yakima Training Center, located to the north and west of the Hanford Site.

In recent years, the number of nesting ferruginous hawks has increased, at least in part because the hawks have accepted steel powerline towers as nesting sites. Only about 50 pairs are believed to be nesting in Washington. Other raptors that nest on the Hanford Site are the prairie falcon, northern harrier, red-tailed hawk, Swainson's hawk, and kestrel. Burrowing owls, great horned owls, barn owls, and long-eared owls also nest on the site but in smaller numbers.

4.9.1.5 Mammals. Approximately 39 species of mammals have been identified on the Hanford Site (Fitzner and Gray 1991), and a complete list can be found in Cushing (1992). The largest vertebrate predator inhabiting the Hanford Site is the coyote, which ranges all across the site. Coyotes have been a major cause of destruction of Canada goose nests on Columbia River islands, especially islands upstream from the abandoned Hanford townsite. Bobcats and badgers also inhabit the Hanford Site in low numbers.

Black-tailed jackrabbits are common on the Hanford Site, mostly associated with mature stands of sagebrush. Cottontails are also common but appear to be more closely associated with the buildings, debris piles, and equipment laydown areas associated with the onsite laboratory and industrial facilities.

Townsend's ground squirrels occur in colonies of various sizes scattered across the Hanford Site but marmots are scarce. The most abundant mammal inhabiting the site is the Great Basin pocket mouse. It occurs all across the Columbia River plain and on the slopes of the surrounding ridges. Other small mammals include the deer mouse, harvest mouse, grasshopper mouse, montane vole, vagrant shrew, and Merriam's shrew.

The Hanford Site has seven species of bats that are known to be or are potential inhabitants, arriving mostly as fall or winter migrants. The pallid bat frequents deserted buildings and is thought to be the most abundant of the various species. Other species include the hoary bat, silver-haired bat, California brown bat, little brown bat, Yuma brown bat, and Pacific western big-eared bat.

A herd of Rocky Mountain elk is present on the ALE Reserve. It is believed these animals immigrated to the reserve from the Cascade Mountains in the early 1970s. This herd had grown from approximately 6 animals in 1972 to 119 animals in the spring of 1992. Elk frequently move off the ALE Reserve to private lands located to the north and west, particularly during late spring, summer, and early fall. However, while the elk are on the Hanford Site, they restrict their activities to the ALE Reserve. Lack of water and the high level of human activity presumably restrict the elk from using other areas of the Hanford Site. Despite the arid climate and their unusual habitat, these elk appear to be very healthy; antler and body size for given age classes are among the highest recorded for this species (McCorquodale et al. 1989). In addition, reproductive output is also among the highest recorded for this species. Elk remain on the ALE Reserve because of the protection it provides from human disturbance.

Mule deer are found throughout the Hanford Site, although areas of highest concentrations are on the ALE Reserve and along the Columbia River. Deer populations on the Hanford

Site appear to be relatively stable. The herd is characterized by a large proportion of very old animals (Eberhardt et al. 1982) and high fawn mortality. Islands in the Hanford Reach of the Columbia River are used extensively as fawning sites by the deer (Eberhardt et al. 1979) and thus are a very important habitat for this species. Hanford Site deer frequently move offsite and are killed by hunters on adjacent public and private lands (Eberhardt et al. 1984).

The ecological survey conducted on an area adjacent to the proposed SNF facility site recorded (by presence or sign) 12 bird, 7 mammal, and 3 reptile species.

4.9.2 Wetlands

Several habitats on the Hanford Site could be considered as wetlands. The largest wetland habitat is the riparian zone bordering the Columbia River. The extent of this zone varies, but it includes extensive stands of willows, grasses, various aquatic macrophytes, and other plants. The zone is extensively impacted by both seasonal water level fluctuations and daily variations related to power generation at Priest Rapids Dam immediately upstream from the site.

Other extensive areas of wetlands can be found within the Saddle Mountain National Wildlife Refuge and the Wahluke Wildlife Refuge Area. These two areas encompass all the lands extending from the north bank of the Columbia River northward to the site boundary and east of the Columbia River down to Ringold Springs. Wetland habitat in these areas consists of fairly large ponds resulting from irrigation runoff. These ponds have extensive stands of cattails (*Typha* sp.) and other emergent aquatic vegetation surrounding the open water regions. They are extensively used as resting sites by waterfowl.

Some wetlands habitat exists in the riparian zones of some of the larger spring streams on the ALE Reserve. These areas are not extensive and usually amount to less than a hectare in size, although the riparian zone along Rattlesnake Springs is probably about 2 kilometers (1.2 miles) in length and consists of peachleaf willows, cattails, and other plants. No wetlands are on or in the vicinity of the proposed project site area.

4.9.3 Aquatic Resources

There are two types of natural aquatic habitats on the Hanford Site: one is the Columbia River, which flows along the northern and eastern edges of the Hanford Site, and the other is provided by the small spring-streams and seeps located mainly in the Rattlesnake Hills. Several artificial water bodies, both ponds and ditches, have been formed as a result of wastewater disposal practices associated with the operation of the reactors and separation facilities. These bodies of water are temporary and will vanish with cessation of activities, but while present, they form established aquatic ecosystems (except West Pond) complete with representative flora and fauna (Emery and McShane 1980). West Pond is created by a rise in the water table in the 200 Areas and is not fed by surface flow; thus, it is alkaline and has a greatly restricted complement of biota.

4.9.3.1 The Columbia River. The Columbia River is the dominant aquatic ecosystem on the Hanford Site and supports a large, diverse community of plankton, benthic invertebrates, fish, and other communities. It is the fifth largest river in North America and has a total length of about 2000 kilometers (~1240 miles) from its origin in British Columbia to its mouth at the Pacific Ocean. The Columbia has been dammed both upstream and downstream from the Hanford Site, and the reach flowing through the area is the last free-flowing, but regulated, reach of the Columbia River in the United States. Plankton populations in the Hanford Reach are influenced by communities that develop in the reservoirs of upstream dams, particularly Priest Rapids Reservoir, and by manipulation of water levels below by dam operations in downstream reservoirs. Phytoplankton and zooplankton populations at Hanford are largely transient, flowing from one reservoir to another. Generally, insufficient time does not allow characteristic endemic groups of phytoplankton and zooplankton to develop in the Hanford Reach. No tributaries enter the Columbia during its passage through the Hanford Site. Gray and Dauble (1977) list 43 species of fish in the Hanford Reach of the Columbia River. Since 1977, the brown bullhead (*Ictalurus nebulosus*) has also been collected, bringing the total number of fish species identified in the Hanford Reach to 44. Of these species, the chinook salmon, sockeye salmon, coho salmon, and steelhead trout use the river as a migration route to and from upstream spawning areas and are of the greatest economic importance. Both the fall chinook salmon and steelhead trout also spawn in the Hanford Reach. The relative contribution of upper river bright stocks to fall chinook salmon runs in the Columbia River increased from about 24 percent of the total in the early 1980s to 50 percent to 60 percent of the total by 1988 (Dauble and Watson 1990). The destruction of other mainstream Columbia spawning grounds by dams has increased the relative importance of the Hanford Reach spawning (Watson 1970, 1973). Fish migrating from the Columbia River up the Snake River would not be expected to pass through the Hanford area because the confluence of the two rivers lies downstream from the Hanford Site.

4.9.3.2 Spring Streams. The small spring streams, such as Rattlesnake and Snively springs, contain diverse biotic communities and are extremely productive (Cushing and Wolf 1984). Dense blooms of watercress occur and are not lost until one of the major flash floods occurs. The aquatic insect production is fairly high as compared to that in mountain streams (Gaines 1987). The macrobenthic biota varies from site to site and is related to the proximity of colonizing insects and other factors.

4.9.4 Threatened, Endangered, and Sensitive Species

Threatened and endangered plants and animals identified on the Hanford Site, as listed by the federal government (50 CFR 17) and Washington (Washington Natural Heritage Program 1994), are shown in Table 4.9-1. No plants or mammals on the federal list of endangered and threatened wildlife and plants (50 CFR 17.11, 17.12) are known to occur on the Hanford Site. However, several species of both plants and animals are under consideration for formal listing by the federal government and Washington.

4.9.4.1 Plants. Four species of plants are included in the Washington listing. Columbia milk-vetch (*Astragalus columbianus* Barneby) and Hoover's desert parsley (*Lomatium tuberosum*) are listed as threatened, and Columbia yellowcress (*Rorippa columbiae* Suksd.) and northern wormwood (*Artemisia campestris* ssp. *borealis* var. *wormskioldii*) are designated as endangered. Columbia milk-vetch occurs on dry land benches along the Columbia River in the vicinity of Priest Rapids Dam, Midway, and Vernita. It also has been found on top of Umtanum Ridge and in Cold Creek Valley near the present vineyards. Hoover's desert parsley grows on steep talus slopes in the vicinity of Priest Rapids Dam, Midway, and Vernita. Yellowcress occurs in the wetted zone of the water's edge along the Columbia River. Northern wormwood is known to occur near Beverley and could inhabit the northern shoreline of the Columbia River across from the 100 Areas.

Table 4.9-1. Threatened (T) and endangered (E) species known or possibly occurring on the Hanford Site.

Common name	Scientific name	Federal	State
Plants			
Columbia milk-vetch	*Astragalus columbianus*		T
Columbia yellowcress	*Rorippa columbiae*		E
Hoover's desert parsley	*Lomatium tuberosum*		T
Northern wormwood	*Artemisia campestris borealis var. wormskioldii*		E
Birds			
Aleutian Canada goose	*Branta canadensis leucopareia*	T	E
Peregrine falcon	*Falco peregrinus*	E	E
Bald eagle	*Haliaeetus leucocephalus*	T	T
White pelican	*Pelecanus erythrorhychos*		E
Sandhill crane	*Grus canadensis*		E
Ferruginous hawk	*Buteo regalis*		T
Mammals			
Pygmy rabbit	*Brachylagus idahoensis*		T
Insects			
Oregon silverspot butterfly	*Speyerra zerene hippolyta*	T	T

4.9.4.2 Animals. The federal government lists the Aleutian Canada goose (*Branta canadensis leucopareia*) and the bald eagle (*Haliaeetus leucocephalus*) as threatened and the peregrine falcon (*Falco peregrinus*) as endangered. In addition to the peregrine falcon, Aleutian Canada goose, and bald eagle, Washington lists the white pelican (*Pelecanus erythrorhynchos*) and sandhill crane (*Grus canadensis*) as endangered and the ferruginous hawk (*Buteo regalis*) as threatened. The peregrine falcon is a casual migrant to the Hanford Site and does not nest here. The Oregon silverspot butterfly (*Speyerra zerene hippolyta*) has recently been classified as a threatened species by both the state and federal governments. The bald eagle is a regular winter resident and forages on dead salmon and waterfowl along the Columbia River; nesting attempts have been made on the Hanford Site, but those have not been successful to date. does not nest on the Hanford Site. Increased use of power poles for nesting sites by the ferruginous hawk on the Hanford Site has been noted. Washington State Bald Eagle Protection Rules were issued in 1986 (WAC-232-12-292). These rules require DOE to prepare a management plan to mitigate eagle disturbance; this has been done by Fitzner and Weiss (DOE/RL 1994). The Endangered Species Act of 1973 also requires that Section 7 consultation be undertaken when any action is taken that may jeopardize the existence of, destroy, or adversely modify habitat of the bald eagle or other endangered species.

Table 4.9-2 lists the designated candidate species that are under consideration for possible addition to the threatened or endangered list. Table 4.9-3 lists the plant species that are of concern in the state of Washington and are presently listed as sensitive or are in one of three monitor groups (Washington Natural Heritage Program 1994).

Sagebrush habitat is considered priority habitat by Washington because of its relative scarcity in the state and its requirement as nesting/breeding habitat by loggerhead shrikes (federal and state candidate species), sage sparrows (state candidate), burrowing owls (state candidate), pygmy rabbits (federal candidate and state threatened), sage thrashers (state candidate), western sage grouse (federal and state candidate), and sagebrush voles (state monitored). Although the last five species were not discovered during the present survey of the proposed SNF site, the habitat should be considered potentially suitable for their use. Pygmy rabbits and western sage grouse have only rarely been seen on the Hanford Site, and then primarily in upland regions. Loggerhead shrikes have been seen frequently on the proposed SNF facility site and are known to select tall big sagebrush as nest sites (Poole 1992). Although this species begins migration at the beginning of August (Poole 1992), one individual was observed during the present survey of the proposed SNF site. However, no nests were located. Ground squirrel burrows used by burrowing owls and owl pellets were observed during the present survey of the proposed SNF site. Numerous sage sparrows were also observed on the proposed SNF site. Pygmy rabbits would not have been observed during this survey because they are primarily crepuscular and nocturnal and may have already begun hibernation. However, this species is not known from lowland portions of the Hanford Site. The closest known ferruginous hawk (federal candidate and state threatened species) nest is approximately 8.9 kilometers (5.3 miles) northwest of the subject area. The subject area should be considered as comprising a portion of the foraging range of this species. No other species listed as endangered or threatened, or candidates for such listing by Washington or federal governments, or species listed as monitor species by Washington State, were observed on the proposed SNF site.

Table 4.9-2. Candidate species.

Common Name	Scientific Name	Federal	State
Mollusks			
Shortfaced lanx	*Fisherola (=Lanx) nuttalli*		X
Columbia pebble snail	*Fluminicola (=Lithoglyphus) columbiana*	X	X
Birds			
Common loon	*Gavia immer*		X
Swainson's hawk	*Buteo swainsoni*		X
Ferruginous hawk	*Buteo regalis*	X	
Western sage grouse	*Centocrcus urophasianus phaios*	X	X
Sage sparrow	*Amphispiza belli*		X
Burrowing owl	*Athene cunicularia*		X
Loggerhead shrike	*Lanius ludovicianus*	X	X
Northern goshawk	*Accipter gentilis*	X	
Harlequin duck	*Histrionicus histrionicus*	X	
Lewis' woodpecker	*Melanerpes lewis*		X
Long-billed curlew	*Numenius americanus*	X	
Sage thrasher	*Oreoscoptes montanus*		X
Flammulated owl	*Otus fammeolus*		X
Western bluebird	*Sialia mexicana*		X
Tricolored blackbird	*Agelaius tricolor*	X	
Golden eagle	*Aquila chrysaetos*		X
Black tern	*Chlidonius niger*	X	
Mammals			
Merriam's shrew	*Sorex merriami*		X
Pacific western big-eared bat	*Plecotus townsendii townsendii*	X	
Pygmy rabbit	*Brachylagus idahoensis*	X	
Insects			
Columbia River tiger beetle	*Cinindela columbica*		X
Plants			
Columbia milk-vetch	*Astragalus columbianus*	X	
Columbia yellowcress	*Rorippa columbiae*	X	
Hoover's desert parsley	*Lomatium tuberosum*	X	
Northern wormwood	*Artemisia campetis borealis* var. *wormskioldii*	X	

Table 4.9-3. Washington plant species of concern occurring on the Hanford Site.

Common Name	Scientific Name	Status[a]
Dense sedge	*Carex densa*	S
Gray cryptantha	*Cryptantha leucophaea*	S
Bristly cyptantha	*Cryptantha interrupta*	S
Shining flatsedge	*Cyperus rivularis*	S
Piper's daisy	*Erigeron piperianus*	S
Southern mudwort	*Limosella acaulis*	S
False-pimpernel	*Lindernia anagallidea*	S
Dwarf desert primrose	*Oenothera pygmaea*	S
Desert dodder	*Cuscuta denticulata*	M1
Thompson's sandwort	*Arenaria franklinii v. thompsonii*	M2
Robinson's onion	*Allium robinsonii*	M3
Columbia River mugwort	*Artemisia lindleyana*	M3
Stalked-pod milkvetch	*Astragalus sclerocarpus*	M3
Medick milkvetch	*Astragalus speirocarpus*	M3
Crouching milkvetch	*Astragalus succumbens*	M3
Rosy balsamroot	*Balsamorhiza rosea*	M3
Palouse thistle	*Cirsium brevifolium*	M3
Smooth cliffbrake	*Pellaea glabella*	M3
Fuzzy beardtongue penstemon	*Penstemon eriantherus*	M3
Squill onion	*Allium scillioides*	M3

The following species may inhabit the Hanford Site, but have not been recently collected, and the known collections are questionable in terms of locations or identification.

Palouse milkvetch	*Astragalus arrectus*	S
Few-flowered blue-eyed Mary	*Collinsia sparsiflora*	S
Coyote tobacco	*Nicotiana attenuata*	S

a. Abbreviations: S, sensitive; taxa vulnerable or declining, and could become endangered or threatened without active management or removal of threats. M1, Monitor group 1; taxa for which there are insufficient data to support listing as threatened, endangered, or sensitive. M2, Monitor group 2; taxa with unresolved taxonomic questions. M3, Monitor group 3; taxa that are more abundant or less threatened than previously assumed.

4.9.5 Radionuclide Levels in Biological Resources

Samples of vegetation and wildlife are routinely collected as part of the site environmental monitoring program and analyzed for various radionuclides. The following summarizes the levels reported in Woodruff and Hanf (1993).

A single sample of vegetation collected on the Hanford Site contained 0.015 picocuries strontium-90 per gram dry weight and 0.0059 picocuries cesium-137 per gram dry weight. These values are lower by nearly an order of magnitude from those reported for the previous five years. Mean values of cesium-137 in upland gamebird muscle (n = 4) in 1992 were 0.02 picocuries per gram wet weight and were about an order of magnitude higher than similar samples collected off of the Hanford Site the previous five years (n = 42). Mean values of cesium-137 in rabbit muscle (n = 12) were 0.09 picocuries per gram wet weight and exceed those collected on the Hanford Site the previous five years (n = 27) by about threefold, and were an order of magnitude higher than samples collected off of the Hanford Site. Values for strontium-90 in rabbit bone (n = 12) had a mean value of 4.08 picocuries per gram wet weight; mean values collected on the Hanford Site for the previous five years (n = 37) were 43 picocuries per gram wet weight, an order of magnitude higher. Mean strontium-90 concentrations in the bones of rabbits (n = 20) collected off of the Hanford Site were 0.37 picocuries per gram wet weight. One sample of muscle collected from a deer in the 200-Areas contained 0.006 picocuries cesium-137 per gram wet weight, nearly two orders of magnitude less than a similar sample collected off of the Hanford Site. Fish populations are safe for human consumption. Radionuclide levels of fish from the Hanford Reach are not significantly higher than those of fish found upstream. Because the confluence of the Snake and Columbia Rivers is downstream from the Hanford Site, the Snake River salmon runs do not migrate through the Hanford reach.

4.10 Noise

Noise is technically defined as sound waves perceptible to the human ear. Sound waves are characterized by frequency, measured in Hertz (Hz), and sound pressure expressed as decibels (dB). Noise levels are often reported as the equivalent sound level (Leq), which normally refers to the equivalent continuous sound level for an intermittent sound, such as traffic noise. The Leq is expressed in A-weighted decibels (dBA) over a specified period of time and is a frequency-weighted measure of sound level related to human hearing characteristics and the concept of equal loudness.

4.10.1 Hanford Site Sound Levels

Most industrial facilities on the Hanford Site are located far enough away from the site boundary that noise levels at the boundary are not measurable or are barely distinguishable from background noise levels. Modeling of environmental noises has been performed for commercial reactors and State Highway 240 through the Hanford Site. These data are not concerned with background levels of noise and are not reviewed here. Two studies of environmental noise were done at Hanford, as described in subsections 4.10.2 and 4.10.3. One study reported environmental noise measurements taken in 1981 during site characterization of the Skagit/Hanford Nuclear Power Plant Site (NRC 1982). The second was a series of site characterization studies performed in 1987 that included measurement of background environmental noise levels at five places on the Hanford Site. Additionally, such activities as well drilling and sampling have the potential for producing noise in the field apart from major permanent facilities. Noise can be disruptive to wildlife and studies have been done to compile noise data in remote areas.

4.10.2 Skagit/Hanford Data

Preconstruction measurements of environmental noise were taken in June 1981 on the Hanford Site (NRC 1982). Monitoring was conducted at 15 sites, showing point noise level reading ranging from 30 to 60.5 dBA. The corresponding values for more isolated areas ranged from 30 to 38.8 dBA. Measurements taken in the vicinity of the sites where the Washington Public Power Supply System was constructing nuclear power plants ranged from 50.6 to 64 dBA, reflecting operation of construction equipment. Measurements taken along the Columbia River near the intake structures for WNP-2 were 47.7 and 52.1 dBA, compared to more remote river noise levels of 45.9 dBA (measured about three miles upstream of the intake structures). Community noise levels from point measurements in North Richland (3000 Area at Horn Rapids Road and Stevens Road [Route 240]) were 60.5 dBA, largely attributed to traffic. North Richland is about 20 miles from the proposed site for SNF facilities.

4.10.3 Basalt Waste Isolation Project Data

Background noise levels were determined at five sites located within the Hanford Site. Noise levels are expressed as equivalent sound levels for 24 hours (Leq-24). The average noise level for these five sites was 38.8 dBA on the dates tested. Wind was identified as the primary contributor to background noise levels with winds exceeding 12 mph significantly affecting noise levels. This study concluded that background noise levels in undeveloped areas at Hanford can best be described as a mean Leq-24 of 24 to 36 dBA (Cushing 1992). Periods of high wind, which normally occur in the spring, would elevate background noise levels.

4.10.4 Noise Levels of Hanford Field Activities

In the interest of protecting Hanford workers and complying with Occupational Safety and Health Administration (OSHA) standards for noise in the workplace, the Hanford Environmental Health Foundation has monitored noise levels resulting from several routine operations performed in the field at Hanford. These included well drilling, pile driving, compressor operations, and water wagon operation. Occupational sources of noise propagated in the field from outdoor activities ranged from 93.4 to 96 dBA.

4.10.5 Noise Related to the Spent Nuclear Fuel Facility

Ambient noise levels at the proposed project SNF site just west of the 200-East Area on the Hanford Site are very low and would be expected to be less than 40 dBAs. The land is currently vacant, and no vehicular traffic transverses the site. A lightly used road borders the eastern side of the proposed SNF site and occasional traffic generates moderate amounts of vehicular noise, but only for those personnel near the road. Existing traffic noise on the Hanford Site is centered primarily on the main arteries leading into the site. These are Route 4 South, which connects with the Richland Bypass (Route 240) and eventually with Interstate 182. Another main road is Route 10, which also connects with Route 240 and leads into the 200 Areas in the site center. It is estimated that 3,300 privately owned vehicles travel to and from the site each day using these roads. The vast majority of the privately owned vehicle movement occurs during the rush hours of 6 to 8 a.m. and 3:30 to 6 p.m. In addition, it is estimated that 3,600 oncoming truck shipments, 445 oncoming rail shipments, and 837 intrasite truck shipments occur each day on the Hanford Site. The movement of all this vehicular traffic generates noise along these affected road corridors. However, little, if any, population exists along these roadways because of the geographic remoteness of work areas on the Hanford Site. Information on noise contours generated by peak rush hour traffic in terms of community Leqs and dBAs is not available at this time.

4.10.6 Background Information

Studies at Hanford of noise propagation have been concerned primarily with occupational noise at work sites. Environmental noise levels have not been extensively evaluated due to the remoteness of most Hanford activities and their isolation from receptors that are covered by federal or state statutes. The Noise Control Act of 1972 and its subsequent amendments (Quiet Communities Act of 1978, 42 USC 4901-4918, 40 CFR 201-211) empower the state to direct. The State of Washington has adopted RCW 70.107, which authorizes the Washington Department of Ecology to implement rules consistent with federal noise control legislation. The Hanford Site is currently in compliance with state and federal noise regulations.

4.11 Traffic and Transportation

4.11.1 Regional Infrastructure

This section discusses the existing transportation environment at and around the Hanford Site. Personnel and most material shipments are transported by road. Bulk materials or large items are shipped by barge. Rail transportation is used only to move irradiated fuel, certain high-level radioactive solid wastes, equipment, and materials (primarily coal). High-level and low-level wastes from spent fuel stabilization are transported to waste management facilities by pipeline.

The regional transportation network in the Hanford vicinity includes the areas in Benton and Franklin Counties from which 93 percent of the commuter traffic associated with the site originates. Interstate highways that serve the area are I-82, I-182, and I-90 (Figure 4-19). Interstate-82 is 8 kilometers (5 miles) south-southwest of the site. Interstate-182, a 24-kilometer (15-mile) long urban connector route 8 kilometers (5 miles) south-southeast of the site, provides an east-west corridor linking I-82 to the Tri-Cities area. Interstate-90 (not shown in Figure 4-19), located north of the site, is the major link to Seattle and Spokane and extends to the east coast; SR 224 (not shown in Figure 4-19), also south of the site, serves as a 16-kilometers

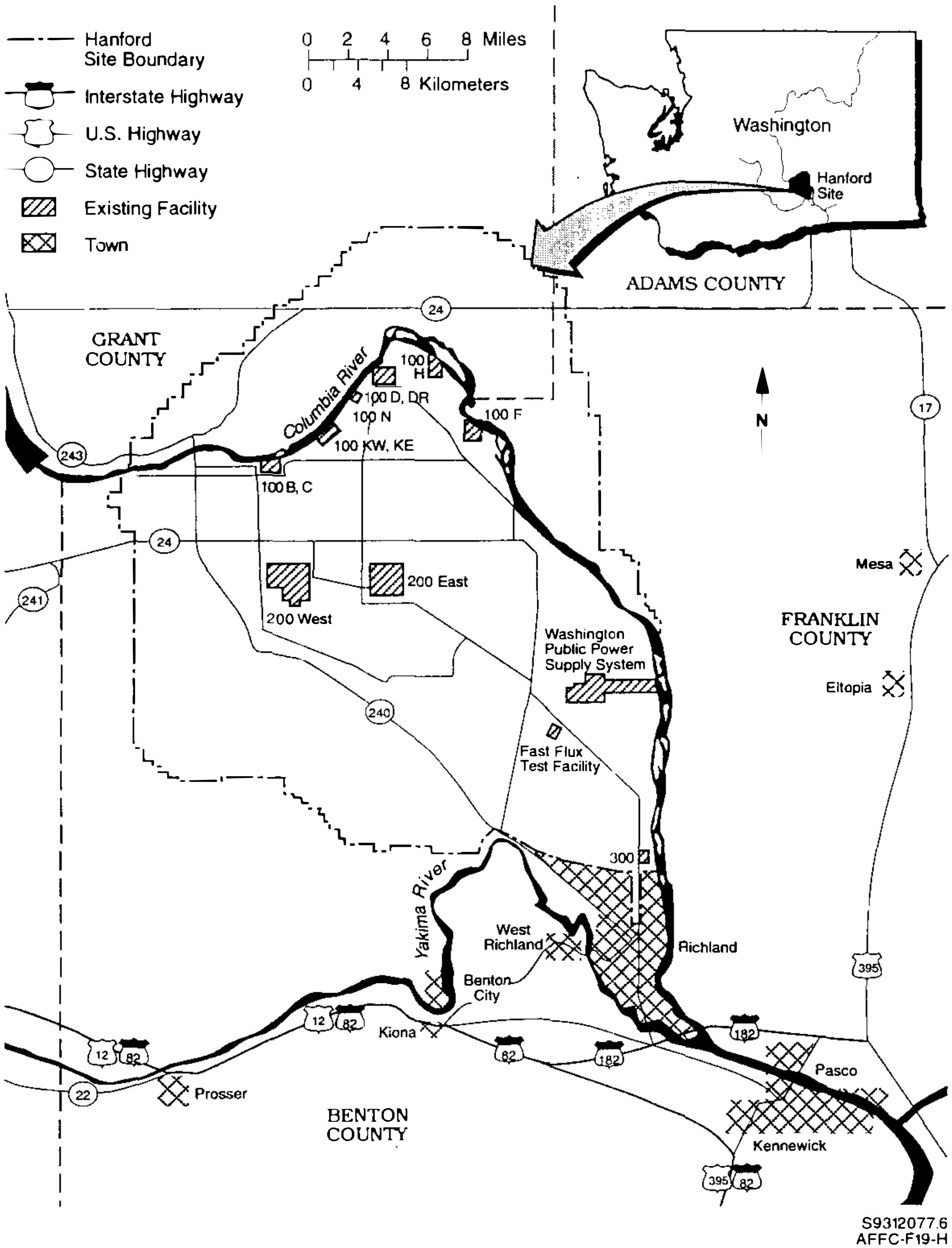

Figure 4-19. Transportation routes in the Hanford vicinity.

(10-mile) link between I-82 and SR 240. State Route 243 exits the northwestern boundary of the site and serves as a primary link between Hanford and I-90. State Route 24 enters the site from the west, continues eastward across the northernmost portion of the site, and intersects SR 17 approximately 24 kilometers (15 miles) east of the site boundary. State Route 17 is a north-south route that links I-90 to the Tri-Cities and joins U.S. Route 395, which continues south through the Tri-Cities. State Route 14 (not shown in Figure 4-19) connects with I-90 at Vantage, Washington, and provides ready access to I-84 (not shown in Figure 4-19) at several locations along the Oregon and Washington border.

General weight, width, and speed limits have been established for highways in the Hanford vicinity. However, no unusual laws or restrictions that have been identified would significantly influence general regional transportation.

Airline passenger and air freight service is provided at the Tri-Cities Airport owned and operated by the Port of Pasco, at Pasco, Washington. The air terminal is located approximately 16 kilometers (10 miles) from the Hanford Site. Delta Airlines provides domestic Boeing-737 and 727 service to Salt Lake City where multiple major airline service is available for domestic and international travel. Two feeder airlines service the Tri-Cities: United Express, a subsidiary of United Airlines, and Horizon Airlines, a subsidiary of Alaska Airlines, provide service to Seattle, Portland, and several other regional cities. Federal Express serves the Tri-Cities by charter airplane from Spokane to Pasco and Airborne Express serves the Tri-Cities with charter airplane from Seattle to the Richland airport, Richland, Washington.

4.11.2 Hanford Site Infrastructure

Hanford's onsite road network consists of rural arterial routes (see Figure 4-20). Only 104 of the 461 kilometers (65 of the 288 miles) of paved roads at Hanford are accessible to the public. Most onsite employee travel occurs along Route 4, with controlled access at the Yakima and Wye barricades. State Route 240 is the main public route through the site. Public highways SR 24 and SR 243 also traverse the site.

The highway network is in excellent condition. A recently completed major highway improvement project involved repavement and widening of the four-lane access route to the Wye Barricade. The highway network has been used extensively for transporting large

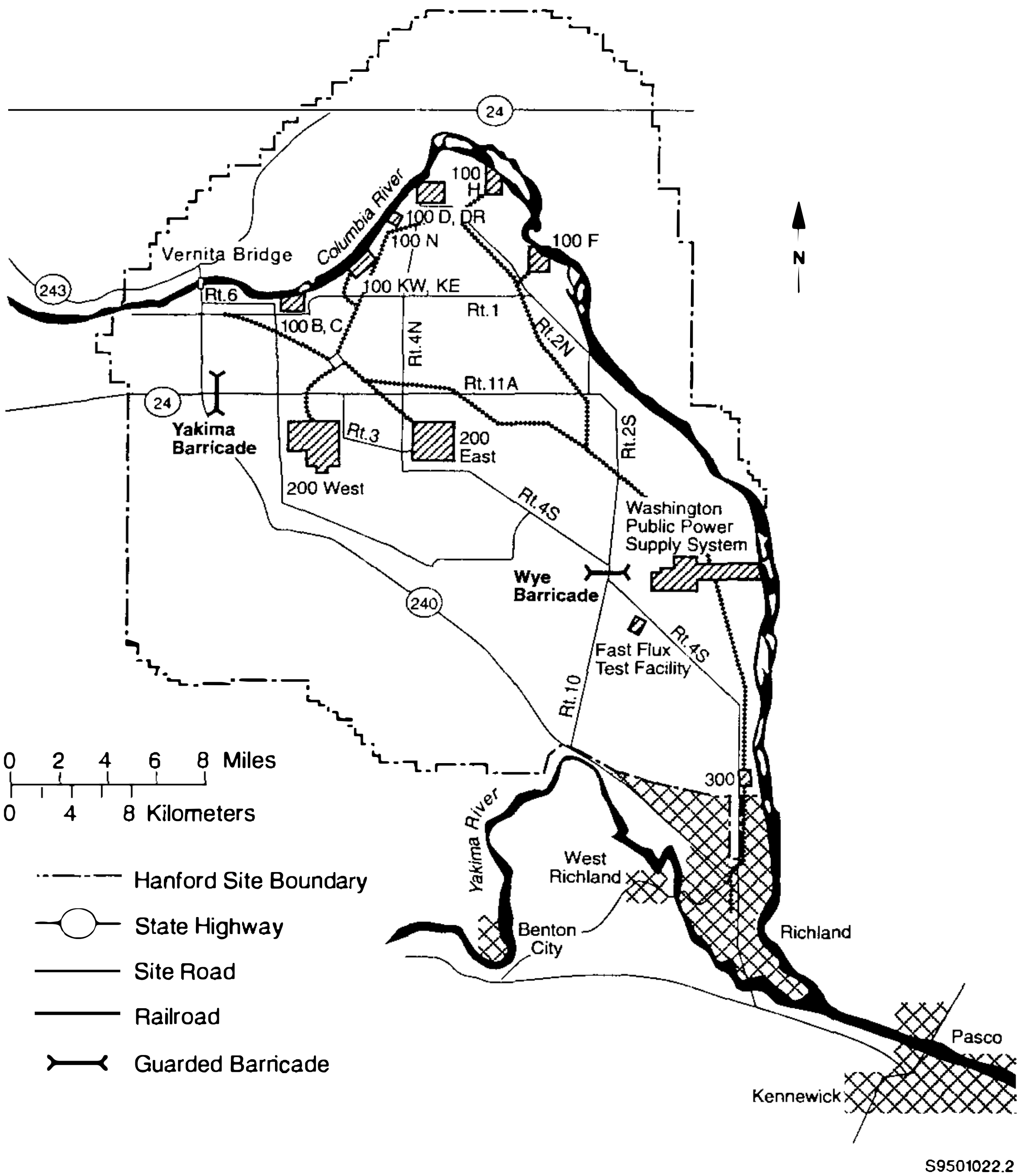

Figure 4-20. Transportation routes on the Hanford Site.

equipment items, construction materials, and radioactive materials. Resurfacing, sealing, and restoration programs are currently planned for segments of SR 17, SR 224, SR 240, and U.S. Route 395.

In 1988 about 32 percent of the work force at Hanford worked in offices in Richland. The remaining work force was on the site. Approximately 80 percent of the work force resides in the Tri-Cities: Richland (45 percent), Kennewick (28 percent), and Pasco (7 percent). Approximately 1600 of the employees on the site use bus transportation.

In 1988 nearly 12 million miles were logged by DOE vehicles at Hanford. In addition, an estimated 3,300 privately owned vehicles were driven onsite each weekday and 560 were driven onsite each weekend day. Assuming a round-trip distance of 30 miles onsite for each of these vehicles, a total of about 40 million miles were driven annually by workers onsite.

The primary highways used by commuters are SR 24, SR 240, and I-182; 10, 90, and 10 percent of the work force use these routes, respectively (totals to more than 100 percent because some commuters use two of the routes). With these commuting patterns, workers annually travel about 27 million miles offsite. Trucks used for material shipment to Hanford compose about 5 percent of the vehicular traffic on and around the site. At present there are periods of moderate traffic congestion, some of which is expected to be alleviated by a new road to the 200 Areas.

During 1988, 169 accidents were reported onsite, with 20 involving DOE vehicles. The other accidents involved privately owned vehicles and included seven injury accidents and one fatal accident on SR 240. Among offsite highway segments of concern, most accidents occurred along I-82. According to available data, the 15 accidents involving trucks in 1987 in the Benton/ Franklin county study area resulted in 13 injuries and 3 fatalities.

Onsite rail transport is provided by a short-line railroad owned and operated by DOE. This line connects just south of the Yakima River with the Union Pacific line, which in turn interchanges with the Washington Central and Burlington Northern railroads at Kennewick. AMTRAK passenger rail service is provided in the Tri-Cities at the Burlington Northern depot at Pasco. Approximately 145,000 rail miles were logged at Hanford in 1988, primarily transporting coal to steam plants. Two noninjury rail accidents occurred at Hanford in 1988.

The Hanford Site infrequently uses the Port of Benton dock facilities on the Columbia River for off-loading large shipments. Overland wheeled trailers are then used to transport those shipments to the site. No barge accidents were reported in 1988.

4.12 Occupational and Public Health and Safety

This section summarizes the Hanford Site programs designed to protect the health and safety of workers and the public. It also describes existing radiological and nonradiological conditions and provides a historical perspective on worker and public exposures and potential health effects.

The section is based on existing documentation and generic descriptions. Reference is made to policies, orders, guidance documents, annual occupational exposure and environmental reports, and to other site descriptive documents. The parameters of greatest interest are the history of radiological releases and worker radiation doses, particularly those associated with the storage of SNF.

The DOE, the DOE-RL, and all Hanford Site contractors have established policies to help ensure a safe and healthful workplace for all employees and visitors and to protect the environment and public health and safety. The DOE-RL manager has the overall responsibility for safety and health at the Hanford Site. Each contractor develops and enforces occupational and public health and safety programs that meet or exceed the requirements of DOE orders, other federal agencies, and Washington State.

4.12.1 Occupational Health and Safety

Programs are in place at the Hanford Site to protect workers from radiological and nonradiological hazards. Radiological protection (health physics) programs are based on requirements in regulations and DOE orders, and on guidance in radiological control manuals. Occupational nonradiological health and safety programs are composed of industrial hygiene programs and occupational safety programs.

4.12.1.1 Radiological Health and Safety/Health Physics Program. In order to help ensure that workers at DOE facilities are adequately protected from ionizing radiation, the

DOE promulgates radiation protection standards for occupational workers. These standards include radiation dose limits to control worker dose from both external radiation and internally deposited radionuclides. The current radiation dose limits were promulgated in 10 CFR Part 835, "Occupational Radiation Protection," which was enacted in 1993. This regulation includes limits on total effective dose equivalent to workers, dose to individual organs, and dose to members of the public (including minors and unborn children of workers) that may be incidentally exposed while at DOE facilities.

Hanford contractors base their radiological protection programs, procedures, and manuals primarily on 10 CFR Part 835. This regulation establishes the criteria for radiation protection for occupational workers. It lists allowable doses, establishes a policy on keeping doses as low as reasonably achievable, and specifies training requirements for radiation protection personnel and other workers. The DOE Radiological Control Manual, DOE/EH-0256T, issued by DOE Headquarters, establishes practices for conducting radiological control activities at all DOE sites. The DOE requires monitoring and reporting of radiation exposure records for individual workers and certain visitors. Monitoring is required by 10 CFR Part 835 when the potential exists for an individual to receive an annual effective dose equivalent above 100 millirem (1 millisievert), or an annual dose equivalent to an individual organ greater than 10 percent of DOE occupational exposure limits. Personnel to be monitored are assigned a thermo-luminescent dosimeter that is worn at all times during radiation work on the Hanford Site. This instrument measures the amount and type of external radiation dose the worker receives. Dosimeters for all DOE and contractor personnel are processed by Pacific Northwest Laboratory. The centralized operational dosimetry program reads, records, and summarizes results of dosimetry data as required. Records of occupational exposure are maintained, and reports of radiation dose are provided annually to each worker. Summary reports are also provided to DOE and published periodically (Smith et al. 1992)

4.12.1.2 Radiation Doses to Workers. The reported cumulative doses to all Hanford Site workers and visitors for all activities are given as a baseline for site operations.

In 1993, about 14,500 workers were monitored at the Hanford Site. Of those monitored, 11,000 were classified as radiation workers, with an average annual dose equivalent of 0.02 rem per individual (Lyon). This dose is well below the 10 CFR Part 835 dose limit of 5 rem per year and the DOE Administrative Control Level of 2 rem per year for occupational exposure.

For 1993, the estimated collective dose-equivalent was 200 person-rem for all Hanford Site radiation workers. Based on standard dose-to-health effects conversion factors (ICRP 1991), no health effects would be expected to result among workers so exposed.

The worker radiation dose of most interest in this document is the cumulative collective dose to SNF workers, which is described in the following subsection. The SNF management alternatives considered in this document are similar to those current work activities associated with maintenance and storage of SNF at the Hanford Site.

4.12.1.3 Radiation Dose to K-Basin Workers. On the Hanford Site the bulk of the SNF is stored in the 105-KE and 105-KW Basins, which are collectively referred to as the K-Basins. The K-Basins are located within the 100-K Area of the Hanford Site. The basins are filled with recirculating water to cool the fuel and to provide radiological shielding for personnel working in the facility. Westinghouse Hanford Company (WHC) operates the K Basins for DOE. Therefore the best measure of radiation dose from SNF is the dose to WHC employees assigned to work at the K Basins. The collective radiation dose to WHC K Basin workers over the 2-year period 1991 and 1992 averaged 22 person-rem per year, or approximately 0.4 rem per year for each worker. An average of 58 workers were assigned to the K-Basin during 1991 and 1992, or approximately 29 workers per basin (Holloman and Motzco 1992, 1993).

The nominal collective radiation dose per year of operation of each SNF basin in the 100-K Area is estimated to be 11 person-rem. During the plutonium production mission, each reactor at the Hanford Site had a similar nuclear fuel storage basin associated with its operation. This resulted in an estimated total radiation dose of 2000 person-rem, assuming 179 total operating reactor years plus six years of K-Basin operation following shutdown of the production reactors (Bergsman 1994). Therefore, operation of nuclear fuel storage basins has accounted for approximately 2.4 percent of the total radiological dose received by all Hanford Site workers from 1945 through 1985, 86,100 rem (Gilbert et al. 1993). Based on standard dose-to-health effects conversion factors (ICRP 1991), the dose to SNF workers since Hanford start up would statistically relate to one fatal cancer among these workers.

4.12.1.4 Worker Safety and Accidents. No incidents of overexposure to radiation have been reported to DOE during 1990 and 1991 in association with SNF storage activities at the Hanford Site. Overexposures are defined as any exposure over regulatory limits established by the DOE (WHC 1990; Lansing et al. 1992). In the four-year period from 1991 through 1994, industrial-type accidents resulted in 98 lost working days at the K Basins out of a total of approximately 70,000 days worked.

4.12.1.5 Industrial Hygiene Program. Occupational nonradiological health and safety programs at Hanford are composed of industrial hygiene and occupational safety programs. Industrial hygiene programs address such subjects as toxic chemicals and physical agents, carcinogens, noise, biological hazards, lasers, asbestos, and ergonomic factors. Occupational safety programs address such subjects as machine safety, hoisting and rigging, electrical safety, building codes, welding safety, and compressed gas cylinders.

The governing document is DOE 5480.10, "Contractor Industrial Hygiene Program," dated 6-26-85. The DOE-RL implementing procedure for DOE 5480.10 is RLIP 5480.10 "Industrial Hygiene Program," dated 7-30-90. The procedure establishes additional requirements and direction for implementation of an industrial hygiene program for DOE-RL and its contractors. In addition to the program requirements of DOE 5480.10, the RL Industrial Health Program addresses the following subject areas:

(1) Use of respiratory equipment
(2) Asbestos material
(3) Regulated carcinogen or suspect carcinogenic materials
(4) Sanitation
(5) Control of hazardous materials
(6) Filter testing
(7) Hearing conservation
(8) Indoor air quality
(9) Human factors
(10) Hazardous waste site safety/health management.

The responsibilities and authorities of the Occupational Medical Services Contractor (contracted by DOE to Hanford Environmental Health Foundation) of the Industrial Health Program are also described in DOE 5480.10. These are 1) to provide technical industrial health support services, that is, air and water monitoring; 2) to evaluate, recommend, and train workers in the use of respiratory devices, as requested by DOE-RL and its contractors; 3) to provide an industrial health analytical laboratory; 4) to conduct work environment surveys; 5) to support noise abatement and hearing conservation; and 6) to maintain permanent records of personal exposure monitoring data. Hanford Environmental Health Foundation maintains centralized records and provides DOE-RL and its contractors with the results of monitoring efforts.

The RL contractors are required to do the following:

- Conduct an effective program to educate employees on the potential health hazards in their work environment, the control measures, and the protection necessary to reduce those risks to acceptable levels.

- Inform employees of health hazards and the results from monitoring of harmful toxic or physical agents in the work environment, and document this action.

Records are maintained in accordance with DOE 1324.2, DOE 5483.1A, and DOE 5484.1. Contractors of DOE-RL are required to maintain records of employee toxic and physical agent exposure and potential personal exposure data. Contractors of DOE-RL are also required to maintain Hanford Site material safety data sheets.

The DOE requires that as low as reasonably achievable (ALARA) principles for radiological and nonradiological hazardous materials be applied in the preparation of all health and safety plans, and that all such ALARA criteria are followed during the course of the work.

Training requirements consistent with 29 CFR 1910.120 for entry into sites potentially containing toxic or hazardous material are specified by DOE (29 CFR OSHA 1991).

The DOE-RL requires that all work (including preliminary investigation activities) be conducted in such a manner that it conforms to applicable federal and state safety and health standards and that all operating equipment meets all safety and operability standards and requirements.

4.12.2 Public Health and Safety

The DOE has the responsibility under the Atomic Energy Act to establish the necessary standards to protect members of the public from radiation exposures resulting from DOE activities. In addition, Presidential Order 12088, "Federal Compliance with Pollution Control Standards," requires all federal facilities to comply with the legislative acts and regulations relating to the prevention, control, and abatement of environmental pollution. The Hanford Site is also in compliance with EPA's National Emission Standards for Hazardous Air Pollutants for Radionuclides, 40 CFR 61, Subpart H. The EPA offsite air emissions limiting standard is 10 millirem/year effective dose equivalent to the public. The National Primary Drinking Water Regulations of the Safe Drinking Water Act apply to the drinking water supplies at the Hanford Site. Several radionuclides are included in these water standards (40 CFR 141, 142; 56 FR 33050-33127, 1991) For 1993, the *Hanford Site Environmental Report* (Dirkes et al. 1994) relates that the facility is in compliance with these requirements.

4.12.2.1 Environmental Programs. DOE 5400.1, "General Environmental Protection Program," establishes the requirement for environmental protection programs. The *Hanford Site Environmental Report* is prepared annually pursuant to DOE 5400.1 to summarize environmental data that characterize Hanford Site environmental management performance and regulatory compliance status. The most recent report summarizes the status in 1993 of compliance with environmental regulations, describes programs at the Hanford Site, discusses estimates of radiation dose to the public from Hanford activities, and presents information on effluent monitoring and environmental surveillance, including groundwater monitoring (Dirkes et al. 1994). In 1993, environmental programs were conducted at the Hanford Site to restore environmental quality, manage waste, develop appropriate technology for cleanup activities, and study the environment.

4.12.2.2 Environmental Monitoring/Surveillance Information. Environmental monitoring at the Hanford Site consists of effluent monitoring and environmental surveillance, including groundwater monitoring. Effluent monitoring is performed by the operators at the facility or at the point of release to the environment. Environmental surveillance consists of sampling and analyzing environmental media on and off the Hanford Site to detect and quantify potential contaminants and to assess their environmental and human health significance. The annual Hanford Site Environmental Reports (Dirkes et al. 1994) present a summary of this information for the Hanford Site. The Hanford Site operations contractor, Westinghouse Hanford Company, also reports summary data annually on radioactive and nonradioactive materials released into the environment from facilities they manage (WHC 1993a). Several federal and state laws and regulations require the reporting of radioactive and nonradioactive releases. The Hanford Site reports pursuant to the federal Clean Air Act (Diediker et al. 1994) and Clean Water Act.

4.12.2.3 Natural Cancer Incidence. The probability of an American contracting cancer in their lifetime is 340 in 1000 (American Cancer Society 1993), and 20 percent of Americans will die from cancer, an estimated 526,000 cancer deaths in 1993. Table 4.12-1 shows the estimated 1993 cancer incidence for different types of cancer for the United States and for Washington State. For the United States the probability of contracting cancer in 1993 is 4.9 in 1000, and 2.2 in 1000 of dying from that cancer. For Washington State the probability of contracting cancer in 1993 is 3.2 in 1000, and 1.4 in 1000 of dying from that cancer.

The expected survival period for cancer victims has increased as detection and treatment technologies have improved. Currently, 40 percent of the victims of all forms of cancer survive for at least 5 years.

4.12.2.4 Potential Radiation Doses. Potential radiation doses and exposures to members of the public from releases of radionuclides to air and water at the Hanford Site are calculated and reported annually by the Surface Environmental Surveillance Project at the Pacific Northwest Laboratory.

Table 4.12-1. Estimated 1993 cancer incidence and cancer deaths in the United States and the state of Washington for different forms of cancer (American Cancer Society 1993).

	United States[a] 1993		Washington State[b] 1993	
Type of Cancer	Estimated new cases	Estimated deaths	Estimated new cases	Estimated deaths
All types & sites	1,170,000	526,000	14,825	6,350
Female breast	182,000	46,000	3,300	850
Colon & rectum	152,000	57,000	2,400	950
Lung	170,000	149,000	3,100	2,700
Oral	29,800	7,700	500	125
Uterus	44,500	10,100	600	125
Prostate	165,000	35,000	3,300	700
Skin melanoma	32,000	6,800	600	125
Pancreas	27,700	25,000	475	425
Leukemia	29,300	18,600	550	350

a. Total population 250 million.
b. Total population 5 million.

4.12.2.4.1 Maximally Exposed Individual (MEI) Dose. The MEI is defined in the *Hanford Site Environmental Report* as "an hypothetical person who lives at a location and has a lifestyle such that it is unlikely that other members of the public would receive higher radiation doses" (Dirkes et al. 1994). The potential radiation doses to MEI have been published in annual Hanford Site Environmental Reports since 1957. For 1993, the total potential dose (via air and water pathways) to the MEI from Hanford operations was calculated to be 0.03 mrem (Dirkes et al. 1994). Estimates of the potential cumulative Effective Dose Equivalent (EDE) to the MEI from both air and water sources for the 28-year period 1994 through 1972 were reconstructed by the Hanford Environmental Dose Reconstruction (HEDR) Project (TSP 1994).

The highest cumulative dose to an adult resident for the years 1944 through 1972 from pathways associated with releases to the air was 1 rem; almost all of this dose was received during 1945. The highest cumulative dose to an adult resident for the years 1944 through 1971 from pathways associated with releases to the water was 1.5 rem; about one-half of this was received during the period from 1954 through 1964. Thus the total cumulative dose from both air and water releases was about 2.5 rem. For comparison, the dose received by an average resident during this 28-year period from natural background radiation was approximately 9 rem. Radiation doses received by the public from Hanford releases after 1972 were vanishingly small.

The maximum cumulative dose to the thyroid of a small child for the years 1944 through 1951 was estimated to be 240 rad; the majority of this dose was received during 1945.

4.12.2.4.2 Population Dose - Estimates of the potential cumulative dose to the population within 50 miles (80 km) of the Hanford Site for 1944 through 1972 were estimated from the releases to air and water developed by the Hanford Environmental Dose Reconstruction (HEDR) project. Pathways of exposure associated with releases to the air dominated the population doses until after 1954 when their contribution decreased rapidly. The cumulative population dose during 1944 through 1972 was 100,000 person-rem; essentially all of this dose was received through air pathways in 1945. The cumulative population dose during 1944 through 1972 associated with water pathways was estimated to be about 6,000 person-rem; most of this dose was received during the decade between 1954 and 1964.

The total potential radiation dose to the population within 50 miles (80 km) for 1993 was 0.4 person-rem (Dirkes et al. 1994). By comparison, the total dose received in 1993 by this same population was about 110,000 person-rem.

About 50 cancer deaths would be implied by the total public radiation dose from Hanford activities since 1944 using standard dose-to-health-effects conversion factors (ICRP 91). Essentially all of these would have been a result of radiation exposures received during 1945. For perspective, the population within 50 miles (80 km) of the Site would have experienced about 75,000 cancer deaths in 1993 from all causes.

4.13 Site Services

4.13.1 Water Consumption

The principal source of water in the Tri-Cities and the Hanford Site is the Columbia River, from which the water systems of Richland, Pasco, and Kennewick draw a large portion of the average 4.3×10^7 cubic meters (11.38 billion gallons) used in 1991. Each city operates its own supply and treatment system. The Richland water supply system derives about 67 percent of its water from the Columbia River, approximately 15 to 20 percent from a well field in North Richland, and the remaining from groundwater wells. The city of Richland's total usage in 1991 was 2.1×10^7 cubic meters (5.65 billion gallons). This current usage represents approximately 58 percent of the maximum supply capacity. The city of Pasco system also draws from the Columbia River for its water needs; the 1991 estimate of consumption is 1.1×10^7 cubic meters (2.81 billion gallons). The Kennewick system uses two wells and the Columbia River for its supply. These wells serve as the sole source of water between November and March and can provide approximately 62 percent of the total maximum supply of 2.8×10^7 cubic meters (7.3 billion gallons). Total usage of those wells in 1991 was 1.1×10^7 cubic meters (2.92 billion gallons).

4.13.2 Electrical Consumption

Electricity is provided to the Tri-Cities by the Benton County Public Utility District, Benton Rural Electrical Association, Franklin County Public Utility District, and City of Richland Energy Services Department. All the power that these utilities provide in the local area is purchased from the Bonneville Power Administration, a federal power marketing agency. The average rate for residential customers served by the three local utilities is approximately $0.0396 per kilowatt hour. Electrical power for the Hanford Site is purchased wholesale from the Bonneville Power Administration. Energy requirements for the site during FY 1988 exceeded 550 average megawatts.

Natural gas, provided by the Cascade Natural Gas Corporation, serves a small portion of residents, with 4800 residential customers in June 1992.

In the Pacific Northwest, hydropower, and to a lesser extent, coal and nuclear power, constitute the region's electrical generation system. Total generating capacity is about 40,270 megawatts. Approximately 74 percent of the region's installed generating capacity is hydroelectric, which supplies approximately 65 percent of the electricity used by the region. Coal-fired generating capacity is 6,702 megawatts in the region, 16 percent of the region's electrical generating capacity. Two commercial nuclear power plants are in service in the Pacific Northwest, with a 2247-megawatt capacity of 6 percent of the region's generating capacity. Oil and natural gas account for about 3 percent of capacity.

The region's electrical power system, more than any other system in the nation, is dominated by hydropower. On average, the region's hydropower system can produce 16,400 megawatts. Variable precipitation and limited storage capabilities alter the system's output from 12,300 average megawatts under critical water conditions to 20,000 average megawatts in record high water years. The Pacific Northwest system's reliance on hydroelectric power means that it is more constrained by the seasonal variations in peak demand than in meeting momentary peak demand.

Throughout the 1980s, the Northwest had more electric power than it required and was operating with a surplus. This surplus has been exhausted, however, and there is only approximately enough power supplied by the existing system to meet the current electricity needs. Hydropower improvement projects currently under construction in the Northwest include about 150 megawatts of new capacity. The cost and availability of several other resources are currently being studied (Northwest Power Planning Council 1986). Approximate rates for current consumption of electricity, coal, propane, natural gas, and other utilities at the Hanford Site are shown in Table 4.13-1.

4.13.3 Waste Water Disposal

The major incorporated areas of Benton and Franklin counties are served by municipal wastewater treatment systems, whereas the unincorporated areas are served by onsite septic systems. Richland's wastewater treatment system is designed to treat a total capacity of 27 million cubic meters per year (a daily average flow of 8.9 million gallons per day with a peak flow of 44 million gallons per day). In 1991 the system processed an average of 4.83 million gallons per day. The Kennewick system similarly has significant excess capacity, with a treatment capability of 12 million cubic meters per year (8.7 million gallons per day); 1991 usage was 4.8 million gallons per day. Pasco's waste-treatment system processes an average of 2.22 million gallons per day, while the system could treat 4.25 million gallons per day or 16.2 liters per day.

4.14 Materials and Waste Management

This section discusses the management of materials and waste and presents both a historic overview and the current status of the various waste types being generated and stored at the Hanford Site. Regulatory requirements governing the management of these materials and wastes are discussed in Section 2.2.

Table 4.13-1. Approximate consumption of utilities and energy on the Hanford Site (1992).

Energy	Consumption	
Electricity	340,000 megawatt-hours	
Coal	45,000 metric tons	(50,000 tons)
Fuel Oil	83,000 cubic meters	(22,000,000 gallons)
Natural Gas	680,000 cubic meters	(24,000,00 cubic feet)
LPG-propane	110 cubic meters	(29,000 gallons)
Gasoline	3,600 cubic meters	(950,000 gallons)
Diesel	1,700 cubic meters	(450,000 gallons)
Other Utilities		
Water	15,000,000 cubic meters	(4,000+ million gallons)
Power Demand	57 megawatts	

In order for Hanford programs to meet operational and mission requirements, many hazardous materials are or have been used onsite. Hazardous materials are not waste, but when no longer useful, may become waste. Because of the potential for impacts to human health and the environment, hazardous materials have been included in Subsection 4.14.7.

Wastes at the Hanford Site are generated by both facility operations and environmental restoration activities. Facility operations include nuclear and non-nuclear research, materials testing, laboratory analysis, high-level waste stabilization, and nuclear fuel storage, manufacturing, repair and maintenance, and general office work. They also include operation of all waste management facilities for treatment, storage, or disposal of Hanford wastes, as well as any waste shipped to Hanford for storage or disposal. Environmental restoration operations include remediation (identifying and arranging for the cleanup of inactive waste sites) and decontamination and decommissioning of surplus facilities.

Wastes and materials handled at the Hanford Site are described in subsections 4.14.1 through 4.14.7. These wastes and materials have been classified as high-level waste (discussed in detail in subsection 4.14.1), transuranic waste (discussed in detail in subsection 4.14.2), mixed low-level waste (discussed in detail in subsection 4.14.3), low-level waste (discussed in detail in subsection 4.14.4), hazardous waste (discussed in detail in subsection 4.14.5), industrial solid waste (discussed in detail in subsection 4.14.6), and hazardous materials (discussed in detail in subsection 4.14.7). Table 4.14-1 shows expected waste disposal rates as of the year 2000, including the expected disposition.

The total amount of waste generated and disposed of at the Hanford Site has been, and is being, reduced through the efforts of the pollution prevention and waste minimization programs at the site. The Hanford Waste Minimization (and Pollution Prevention) Program is an ambitious program aimed at source reduction, product substitution, recycling, surplus chemical exchange, and waste treatment. The program is tailored to meet Executive Order 12780, DOE orders, RCRA, and EPA guidelines. All wastes on the Hanford Site, including radioactive, mixed, hazardous and non-hazardous regulated wastes are included in the Hanford Waste Minimization Program.

Table 4.14-1. Baseline waste quantities as of the year 2000 at Hanford[a].

Waste identification	Annual disposal volume from stabilization operations wastes (m^3/yr)	Annual disposal volume from stabilization of stored wastes (m^3/yr)	Total annual disposal volume from all waste stabilization (m^3/yr)	Disposition
High-level waste solid[b]	0	240	240[c]	Interim onsite storage[d]
Transuranic waste solid[e]	0	170	170[c]	Interim onsite storage[f]
Low-level waste solid[g]	13,000	7,000	20,000	Onsite disposal
Mixed waste solid[g]	300	0	300	Interim onsite storage
Hazardous waste liquid and solid	100	0	100	Offsite disposal
Other waste				
nonhazardous				
liquid	2,000,000	10,000,000	12,000,000	Liquid effluent
solid	38,000	0	38,000	Onsite disposal
sewage				
liquid[h]	210,000	0	210,000	Liquid effluent
solid[i]	4	0	4	Onsite disposal

a. Baseline values are projected from 1988 data.
b. Liquid high-level waste (HLW) is held in interim storage and then processed to a solid form for disposal.
c. The baseline value is taken from 1988 data for planned future activities.
d. These wastes are targeted for disposal at a federal repository.
e. Liquids containing transuranics are processed as HLW.
f. These wastes are targeted for disposal at WIPP.
g. Solidified or absorbed-liquid-waste quantities are included in the solid waste quantity.
h. Liquid effluents from sewage treatment operations.
i. Solids from sewage treatment operations.

Reductions in the volumes of radioactive wastes generated have been achieved through methods such as intensive surveying, waste segregation, recycling, and use of administration and engineering controls. Some examples of waste reduction follow:

- Waste minimization efforts have reduced the volume of waste water discharged to process trenches in the 300 Area by more than 5,600 cubic meters (>1.5 million gallons) per day. By the end of 1992, waste reduction efforts had reduced liquid waste by more than 22,000 cubic meters (>5.8 million gallons) (Woodruff and Hanf 1993).

- In 1991, 440,645 kilograms (971,440 pounds) of ferrous metals, 49,323 kilograms (108,737 pounds) of nonferrous metals, 275 cubic meters (9,076 cubic feet) of wood scrap, and 136,077 kilograms (299,993 pounds) of scrap paper were recycled. During 1992, approximately 181,440 kilograms (400,000 pounds) of paper were recycled (Woodruff and Hanf 1993).

On-going projects include packaging reduction, waste minimization design, and technology transfer.

Databases are used at the Hanford Site to track and manage waste management information. These databases have been screened to ensure that the information supplied is supported by official databases, reports, or other public documents. Although the most reliable data available have been used to quantify and characterize waste volumes, past waste volumes are imprecise and may be subject to change as characterization of previously disposed waste is undertaken and completed.

4.14.1 High-Level Waste

High-level radioactive waste is defined in the Nuclear Waste Policy Act of 1982 (PL 97-425) as "(A) the highly radioactive material resulting from the reprocessing of SNF, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (B) other highly radioactive material that the [Nuclear Regulatory Commission], consistent with existing law, determines by rule requires permanent isolation."

High-level waste at Hanford was generated from the reprocessing of production reactor fuel for the recovery of plutonium, uranium, and neptunium for defense and other national programs of spent reactor fuel and irradiated targets. Radioactive waste generated on the Hanford Site from 1988 through 1990 is shown in Table 4.14-2.

4.14.1.1 Historic Overview. Until recently, the primary mission of the Hanford Site was production of special nuclear material for defense purposes. Since 1943, the Hanford Site has been involved in fabrication of reactor fuel elements, operation of production reactors,

Table 4.14-2. Radioactive waste generated on the Hanford Site from 1988-1990 in kilograms (excluding mixed waste).

Calendar Year	Low-Level Waste	Transuranic Waste	High-Level Waste
1988	3,800,000	21,900	0
1989	8,300,000	27,200	0
1990	3,600,000	24,500	0

Source: DOE 1991.

processing of irradiated fuel, separation and extraction of plutonium and uranium, preparation of plutonium metal, and decontamination and decommissioning activities. Between 1943 and 1964, 149 single-shell tanks were built to store liquid radioactive wastes. No new wastes have been added to these tanks since 1980; much of the liquid waste originally stored in the single-shell tanks has been transferred to some of the 28 one-million gallon double-shell tanks for safer storage (DOE 1993c).

High-level waste has been accumulating at Hanford since 1944. Most of these high-level wastes have undergone one or more treatment steps (e.g., neutralization, precipitation, decantation, or evaporation) and will eventually require incorporation into a stable, solid medium (e.g., glass) for final disposal (DOE 1993d, 1992b).

Between 1956 and 1990, the Plutonium and Uranium Recovery through EXtraction (PUREX) plant processed irradiated reactor fuel to extract plutonium and uranium (DOE 1982). The wastes from the PUREX process were placed in double-shell tanks after 1970, and are the second high-level waste stream (DOE 1993c).

Cesium and Strontium Capsules: From 1968 to 1985, most of the high-heat emitting nuclides (strontium-90 and cesium-137, plus their daughters) were extracted from the old tank waste, converted to solids (strontium fluoride and cesium chloride), placed in double-walled metal cylinders (capsules) about 50 centimeters (20 inches) in length and 5 centimeters (2 inches) in diameter, which were stored in the Waste Encapsulation and Storage Facility in water-filled pools (DOE 1993d).

4.14.1.2 Current Status. There are two high-level waste streams at Hanford: the single-shell tank wastes and double-shell tank PUREX aging wastes. All wastes contained in double-shell tanks consist of mixtures of high-level wastes, transuranic waste, and several low-level wastes, and are managed as if they contain high-level waste. The single-shell tank wastes make up 95 percent of the Hanford Site high-level mixed waste (DOE 1993c).

There are currently 164,000 cubic meters (214,500 cubic yards) of wastes in the single-shell tanks, which are managed as high-level waste. The waste is multi-phased: most is sludge with interstitial liquids; some is in the form of crystalline solids, and there are some supernatant liquids present in the tanks. There are currently 92,000 cubic meters (120,000 cubic yards) of PUREX wastes in the double-shell tanks (DOE 1992e).

No known treatment is currently possible for these two waste streams, although it is planned to treat high-level wastes in the Hanford Waste Vitrification Plant, for which construction is scheduled to begin in 2002, with an operational start date in 2009 (DOE 1993c).

No high-level wastes are expected to be generated in 1995 from SNF management activities.

Cesium and Strontium Capsules: The total number of cesium capsules produced is 1,577. As of August 19, 1993, the number of known dismantled cesium capsules is 249; these have been put to beneficial use and are not expected to be returned. The total number of remaining capsules requiring disposal is 1,328. Of the 1,328 remaining capsules, 959 are in storage at Hanford, and 369 capsules have been leased for beneficial use. One of these capsules developed a small leak, and others have shown signs of bulging, so current plans are to bring all leased capsules back to the Hanford Site (DOE 1993d).

The total number of strontium capsules produced is 640. As of August 19, 1993, the number of known dismantled strontium capsules is 35; these have been put to beneficial use and are not expected to be returned. The total number of remaining capsules requiring disposal is 605. Of the 605, 601 are in storage at Hanford, and 4 have been leased offsite for beneficial use.

Therefore, at present 1,328 cesium capsules (2.47 cubic meters - 3.23 cubic yards) and 605 strontium capsules (1.08 cubic meters - 1.41 cubic yards) require storage. Nine-hundred and fifty-nine cesium capsules and 605 strontium capsules are stored in pools of water in the Waste Encapsulation and Storage Facility. The capsules will be stored at Hanford until they can be transported to a proposed national repository (DOE 1992d).

4.14.2 Transuranic Waste

Transuranic waste is defined in the Atomic Energy Act of 1954 (42 U.S.C. 2014[ee]) as "material contaminated with elements that have an atomic number greater than 92, including neptunium, plutonium, americium, and curium, and that are in concentrations greater than 10 nanocuries per gram, or in such other concentrations as the Nuclear Regulatory Commission may prescribe to protect the public health and safety."

Transuranic waste is primarily generated by research and development activities, plutonium recovery, weapons manufacturing, environmental restoration, and decontamination and decommissioning. Most transuranic waste exists in solid form (e.g., protective clothing, paper trash, rags, glass, miscellaneous tools, and equipment). Some transuranic waste is in liquid form (sludges) resulting from chemical processing for recovery of plutonium or other transuranic elements.

4.14.2.1 Historic Overview. Prior to 1970 all DOE-generated transuranic waste was disposed of onsite in shallow, unlined trenches. From 1970 to 1986, transuranic wastes were segregated from other waste types and disposed in trenches designated for retrieval. Since 1986 all transuranic waste has been segregated and placed in retrievable storage pending shipment and final disposal in a permanent geologic repository (DOE 1992d, 1993g).

4.14.2.2 Current Status. Currently, all transuranic wastes are stored in above-grade storage facilities in the Hanford Central Waste Complex and Transuranic Waste Storage and Assay Facility. The plan is to ship the stored transuranic waste to the Waste Isolation Pilot Plant near Carlsbad, New Mexico for final disposal. The inventory of transuranic wastes is given in Table 4.14-3.

4.14.3 Mixed Low-Level Waste

Mixed low-level waste is defined as mixtures of low-level radioactive materials and (chemically and/or physically) hazardous wastes. Typically, mixed low-level waste includes a

Table 4.14-3. Transuranic waste inventory through 1991[a].

Disposition of TRU Waste	Mass of TRU Nuclides (kilograms)	Volume (cubic meters)
Buried Waste	346	109,000[b]
Retrievable Storage	480	10,200

a. Source: DOE 1992d, Figures 3.3-3.6.
b. This number includes soils contaminated with TRUs.

variety of contaminated materials, including air filters, cleaning materials, engine oils and grease, paint residues, photographic materials, soils, building materials, and decommissioned plant equipment.

4.14.3.1 Historic Overview. Between 1987 and 1991, 16,745 cubic meters (21,902 cubic yards) of mixed low-level waste were buried at the Hanford Site (between 1944 and 1986, no differentiation was made between low-level and low-level mixed wastes); all buried low-level wastes from that period are reported in subsection 4.14.4). Another 4,225 cubic meters (5,526 cubic yards) of mixed waste has been accumulating in storage in the Central Waste Complex, located in the 200-West Area (DOE 1993d).

The Hanford Site also receives defueled submarine reactor compartments, which are contaminated with PCBs and lead. These compartments are managed as mixed waste. Several compartments are received each year and placed in a trench in the 200-East Area (DOE 1993b).

4.14.3.2 Current Status. In 1992, 56,245 kilograms (124,000 pounds) of mixed low-level waste were generated. The 78 mixed low-level waste streams at Hanford make up 85,000 cubic meters (111,176 cubic yards) of waste (101,314,863 kilograms - 223,361,010 pounds). Ninety-six percent of the total is beta/gamma emitting waste in the form of mostly aqueous liquid in the double-shell tanks. One stream (double-shell tank miscellaneous waste) accounts for 40,000 cubic meters (52,318 cubic yards) of the mixed low-level wastes, and in combination, the double-shell tank Double-Shell Slurry Feed, double-shell tank Complex Concentrate and double-shell tank Double-Shell Slurry make up another 34,500 cubic meters (45,124 cubic yards). Three mixed low-level waste streams related to the 183-H Solar Evaporation Basin cleaning made up 2,500 cubic meters (3,270 cubic yards) of wastes. These inorganic sludge/particulate wastes have been neutralized and treated for packaging (DOE 1993c).

It is expected that of all the mixed low-level wastes at Hanford, 49 percent cannot be treated until the technology is modified or verified. The remaining 51 percent is to be processed through the 242A-Evaporator (a closed system in which distillates are passed through an ion-exchange system to remove cesium) (DOE 1993c).

In 1992, eight defueled submarine reactor compartment disposal packages were received and placed in Trench 94 of the 200-East Area Low-Level Waste Burial Grounds (Woodruff and Hanf 1993). The Naval Nuclear Propulsion Program will prepare an EIS for their proposal to bury additional reactor compartments at Hanford. As of November 1993, there were a total of 35 submarine reactor compartments stored in Trench 94.

Mixed low-level wastes generated in 1995 from SNF management activities will total 0.4 cubic meters (0.6 cubic yards).

4.14.4 Low-Level Waste

Low-level radioactive waste is defined in the Nuclear Waste Policy Act of 1982 (PL 97-425) as "radioactive material that (A) is not high-level radioactive waste, spent nuclear fuel, transuranic waste, or by-product material...; and (B) the [Nuclear Regulatory Commission], consistent with existing law, classifies as low-level radioactive waste." By-product material is defined in the Atomic Energy Act of 1954 [42 U.S.C. 2014(e)(2)] as "(1) any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material, and (2) the tailings or wastes produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content."

Commercial fuel low-level waste can be generated by fuel fabrication and reactor operations. Low-level waste also results from commercial operations by private organizations that are licensed to use radioactive materials. These include institutions engaged in research and various medical and industrial activities. Some low-level waste is also generated by DOE environmental restoration activities. Other low-level wastes will be generated in future years by routine decommissioning and decontamination operations.

4.14.4.1 Historic Overview. From 1944 to 1991, approximately 558,916 cubic meters (731,034 cubic yards) of low-level waste was buried at Hanford (DOE 1993d). Between 1944 and 1986, no differentiation was made between low-level and low-level mixed wastes - all data from that period are reported in this section. Another 130 cubic meters (170 cubic yards) was placed into storage.

U.S. Ecology operates a licensed commercial low-level waste burial ground at Hanford on a site that is leased to the State of Washington. Although physically located on the Hanford Site, it is not considered part of the Hanford facility. The site area is 40 hectares (99 acres), of which 29.5 hectares (72.9 acres) is considered usable, with 11.9 hectares (29.4 acres) used by the end of 1991. Through 1991 338,500 cubic meters (442,741 cubic yards) of low-level wastes had been disposed of at this site (DOE 1992d).

4.14.4.2 Current Status. Solid low-level waste currently is placed in unlined, near-surface trenches at the 200-Area Low-Level Waste Burial Grounds. Onsite sources at the Hanford Site generated about 4500 square meters of low-level waste in 1992. Table 4.14-4 lists quantities of radioactive materials received at the Hanford Site from offsite generators over 5 years. The site continues to receive low-level waste from offsite generators for disposal. Major sources of this waste have been the Puget Sound Naval Shipyard in Washington, Brookhaven National Laboratory in New York, and Lawrence Berkeley Laboratory in California. Other points of origin include DOE facilities at nuclear power stations in Shippingport, Pennsylvania; Bechtel in Albany, Oregon; and Wood River in Charleston, Rhode Island (DOE 1993d). The U.S. Ecology commercial low-level burial ground continues to operate.

Table 4.14-4. Offsite low-level waste receipts summary (from 1987 through 1991).[a]

Year	Volume (m^3)	Activity (curies)
1987	7,000	68,000
1988	5,000	107,000
1989	600	1,500
1990	5,500	240,000
1991	5,300	489,000

a. Source: Draft Environmental Restoration and Waste Management Fiscal Year 1993 Site-Specific Plan for the Richland Field Office (DOE 1993d). (Does not include waste quantities received at the U.S. Ecology low-level burial ground.)

In 1995, 174.5 cubic meters (228.3 cubic yards) of low-level wastes will be generated from SNF management activities. Of this amount, 167.2 cubic meters (218.7 cubic yards) are contact handled, and 7.3 cubic meters (9.6 cubic yards) are remote handled.

4.14.5 Hazardous Waste

Hazardous waste is defined in the State of Washington Dangerous Waste Regulations (WAC 173-303) as solid waste designated by 40 CFR Part 261 and regulated as hazardous wastes by the EPA. The State of Washington designates wastes as either "dangerous waste" or "extremely hazardous waste." Hazardous wastes are generated during normal facility operations and environmental restoration activities at the Hanford Site (Table 4.14-5).

Mixed wastes are wastes that contain both hazardous waste (regulated under the Resource Conservation and Recovery Act) and radioactive waste (regulated under the Atomic Energy Act). The following special nuclear material production and site restoration activities have generated or may generate mixed waste:

- fabrication of reactor fuel elements
- operation of the production reactors
- processing of irradiated fuel
- separation and extraction of plutonium and uranium
- preparation of plutonium metal
- environmental restoration (i.e., soil and groundwater cleanup)
- research and development support projects
- maintenance and operations support.

Table 4.14-5. Hazardous waste generated on the Hanford Site from 1988 through 1992 (including mixed waste).

Calendar year	Hazardous waste (t)	Mixed waste (t)	Total (t)
1988	80,000	25,000	105,000
1989	66,000	9400	75,000
1990	780	12,000	13,000
1991	330	4600	4900
1992	620	3400	4000

Tank wastes constitute 99 percent of the mixed wastes at the Hanford Site. The Hanford Site currently has 233,689 cubic meters (305,654 cubic yards) of mixed wastes stored in these tanks: 145,952 cubic meters (190,898 cubic yards) of high-level waste, 3,935 cubic meters (5,147 cubic yards) of mixed transuranic waste, and 84,802 cubic meters (110,917 cubic yards) of mixed low-level waste. These wastes consist of 108 different waste streams (2 high-level waste, 22 mixed transuranic waste, and 84 mixed low-level waste). Of the 108 identified waste streams, 97 are still being generated. Additional environmental restoration waste streams are expected. Their numbers and types remain to be determined (DOE 1993c).

The Resource Conservation and Recovery Act components of mixed waste at the Hanford Site are mainly the following listed wastes: D002B (alkaline liquids, 22 streams), D006B (cadmium, 29 streams), D007 (chromium, 34 streams), D008B (lead, 30 streams), and F003 (nonchlorinated solvents, 30 streams). Waste sources are primarily the separations and extraction processes that were used to produce special nuclear material (DOE 1993c).

4.14.5.1 Historic Overview. In the past, hazardous waste generated at Hanford was either shipped offsite, recycled, or treated onsite. Hazardous waste was also disposed of onsite (e.g., buried in trenches, burial grounds, or discharged to cribs or directly to the soil). For example, from 1943 through 1945, acids from a pipe-cleaning operation were discharged to the soil through two side-by-side cribs in an area west of the old White Bluffs townsite. From 1955 through 1973, approximately 379-2,271 cubic meters (100,000-600,000 gallons) of organic liquids, including carbon tetrachloride, were discharged to the soil in the 200-West Area. Drums containing approximately 19 cubic meters (5,000 gallons) of organic solvent (primarily hexone) were buried at the 618-9 burial ground north of the 300 Area. Many of these disposal sites have been or will be closed under RCRA or remediated under CERCLA (DOE 1993d).

4.14.5.2 Current Status. As of March 15, 1993, the Hanford Site contained 64 interim status treatment, storage, or disposal units. Present plans are that final RCRA permits will be sought for 24 of these 64 interim status treatment, storage, or disposal units. Thirty-four units will be closed under interim status. Six units will be dispositioned through other regulatory options. Future circumstances may cause these numbers to change. The treatment, storage, or disposal units within the Hanford facility include, but are not limited to, tank systems, surface impoundments, container storage areas, waste piles, landfills, and miscellaneous units. Other RCRA permits, such as research, development, and demonstration permits (for example, the 200-Area Liquid Effluent Treatment Facility), are also being pursued (DOE 1993d).

The principal present waste management practice for newly generated nonradioactive hazardous waste is to ship it offsite for treatment, recycling, recovery, and/or disposal. The Nonradioactive Dangerous Waste Storage Facility (616 Building) and the 305-B Waste Storage Facility are the only active facilities storing nonradioactive hazardous waste (other than less than 90-day storage areas) (DOE 1992d, 1993d), other than two boxes (one containing mixed and one containing nonradioactive waste) stored in the 222-S laboratory complex.

Hazardous wastes generated in 1995 from SNF management activities will total 2.2 cubic meters (2.9 cubic yards).

4.14.6 Industrial Solid Waste

Solid wastes are generated in all areas of the Hanford Site. Nondangerous solid wastes include the following nonradioactive, nonhazardous wastes:

(a) construction debris, office trash, cafeteria waste/garbage, empty containers, and packaging materials, medical waste, inert materials, bulky items such as appliances and furniture, solidified filter backwash and sludge from the treatment of river water, failed and broken equipment and tools, air filters, uncontaminated used gloves and other clothing, and certain chemical precipitates such as oxalates

(b) nonradioactive friable asbestos (regulated under the Clean Air Act)

(c) ash generated from powerhouses

(d) nonradioactive demolition debris from decommission projects.

4.14.6.1 Historic Overview. Both prior to and after establishment of the reservation, a number of landfills have been used on the Hanford Site for solid waste disposal, including the Horn Rapids, Central, Original Central, White Bluffs, East White Bluffs, Wahluke Slope and Hanford Townsite Landfills.

The active Hanford Site Solid Waste Landfill, located in the 200-Area, began operation in 1973. Nondangerous wastes in category (a) above are buried in the solid waste section of the Solid Waste Landfill, located in the 200-Area. Nonradioactive friable asbestos is buried in designated areas at the Solid Waste Landfill. The nonradioactive dangerous waste section of the landfill was closed to chemicals in January 1985, and closed to asbestos in May 1988. Ash generated at powerhouses in the 200-East and 200-West Areas is buried in designated sites near those powerhouses. Demolition waste from 100-Area decommissioning projects is buried in situ or in designated sites in the 100 Areas (Woodruff and Hanf 1993; WHC 1993b). Solid waste has also been sent to the City of Richland landfill.

4.14.6.2 Current Status. In 1992, 22,213 cubic meters (29,054 cubic yards) of solid waste and 1,017 cubic meters (1,330 cubic yards) of asbestos were deposited in the solid waste section of the Solid Waste Landfill. Pit 10 was opened for disposal of inert material as defined in Washington Administrative Code (WAC) 173-304, and a total of 11,389 cubic meters (14,986 cubic yards) were disposed of there. A summary of the solid waste disposed of at the Hanford Site from 1973 through 1992 is shown in Table 4.14-6. The landfill is currently scheduled for closure in 1997 (WHC 1993b). Quantities of solid waste disposed of at the City of Richland Landfill are not readily available.

4.14.7 Hazardous Materials

A hazardous chemical is any chemical that poses a physical or health hazard [as defined in 29 CFR 1900.1200(c)]. The Emergency Planning and Community Right-to-Know Act sets forth reporting requirements (Tier 1 and Tier 2) that provide the public with information on hazardous chemicals to enhance community awareness of chemical hazards and facilitate the development of state and local emergency response plans.

Table 4.14-6. 1973-1992: Historical annual volume of onsite buried solid sanitary waste in cubic meters per year.

	Volume (m^3/year)											
Waste Type	73-81	82	83	84	85	86	87	88	87	90	91	92
Construction Debris[a]	4,149	5,819	9,494	10,378	10,789	14,254	14,316	12,842	12,469	10,088	5,666	7,330
Metals[b]	1,383	1,940	3,165	3,459	3,596	4,751	4,772	4,281	4,156	3,363	1,889	2,443
Paper	5,658	7,936	12,946	14,151	14,712	19,437	19,522	17,512	17,003	13,757	7,727	9,996
Miscellaneous[c]	1,383	1,940	3,165	3,459	3,569	4,751	4,772	4,281	4,156	3,363	1,889	2,443
Total	12,573	17,635	28,770	31,447	32,694	43,193	43,382	38,916	37,785	30,571	17,170	22,213

a. Construction Debris: Volume is calculated based on disposal volume (excluding asbestos) at the onsite landfill: Construction debris 33 percent; Metals 11 percent, Paper 45 percent, Miscellaneous Waste 11 percent.
b. Metals: See note b above. Category consists of large bulky items such as appliances and furniture.
c. Miscellaneous: Category includes garbage, packaging, empty containers, medical waste and inert materials.

4.14.7.1 Historic Overview. Hazardous chemicals are used throughout the Hanford Site in facility and environmental restoration operations. The types of chemicals in inventory onsite tend to be static since Hanford's mission involves mainly remediation and decontamination and decommissioning (as opposed to production or processing). The amount of chemicals actually onsite changes from day to day, and there is no requirement to keep a real-time inventory of the quantity of chemicals onsite at any one time. Also, the percentage of hazardous chemicals used onsite that eventually become hazardous waste cannot be determined.

4.14.7.2 Current Status. The Hazardous Materials Inventory Database currently being used to generate Tier 2 data indicates that approximately 1484 hazardous chemicals are reported in inventory at over 783 locations on the Hanford Site. These 1484 chemicals are contained in approximately 2926 different hazardous materials, in weights that range from less than 0.5 kilograms (one pound) to a maximum inventory of 35,658,872 kilograms (78,614,420 pounds).

The DOE has prepared chemical inventory reports required by the Emergency Planning and Community Right-to-Know Act since 1988 (for calendar year 1987). In 1992 the Emergency Planning and Community Right-to-Know Act reporting threshold was exceeded for 53 hazardous chemicals.

5. ENVIRONMENTAL CONSEQUENCES

Descriptions of analyses for various potential environmental consequences as a result of implementing 1) No Action, 2) Decentralization, 3) 1992/1993 Planning Basis, 4) Regionalization, and 5) Centralization Alternatives for interim storage of SNF for the Hanford Site are presented in the following subsections. By and large these discussions are at the programmatic level because in many cases specific alternative treatments and locations, particularly for new facilities, have not been identified for the Hanford Site.

5.1 Overview

An overview of the various alternatives and a brief summary of potential environmental consequences of interest are provided in the following subsections. For purposes of this programmatic analysis, all new facilities were assumed to be constructed in a quarter section of land adjacent to the 200-East Area; commitment of that amount of land within the industrialized 200 Areas would be consistent with the site mission and would not represent a conflict on land use. Up to 15 percent of that area would be disturbed during construction of storage and support facilities where required. A survey of the area described revealed no threatened and endangered species or cultural resources. Routine operations under any of the alternatives would not add significantly to current occupational or near-zero public exposure to radiation. Although not quantified, no significant additions to current releases of criteria pollutants or other hazardous materials would be expected from implementing any of the alternatives. However, such implementation requires a small increase in Hanford's electrical power consumption; the largest increase would be less than 1.5 percent. The influx of workers would probably increase competition for desirable housing and strain teacher/student ratios in some local school districts, the extent of which (although small in any case) would depend on the option chosen.

5.1.1 No Action Alternative

The No Action Alternative identifies the minimum actions deemed necessary for continued safe and secure storage of SNF at the Hanford Site. Upgrade of the existing facilities would not occur other than as required to ensure safety and security. No receipt of fuels from offsite would occur. No research and development would take place; however, characterization of fuel would continue to establish a safety envelope for extended interim storage, fuel would be containerized at the 105-KE Basin, and the first 10 dry storage casks would be procured for FFTF fuel.

Results presented in the Hanford Site Environmental Report for 1992 (Woodruff and Hanf 1993) suggest that under normal conditions no significant environmental effects would be associated with the No Action Alternative. For example, the radiation dose to the maximally exposed individual in the Hanford environs from all Hanford sources was calculated to have been 0.02 mrem and the collective population dose was 0.8 person-rem during 1992. Continued storage of SNF contributed only a small portion of those doses. No health effects would be expected as a result of such small doses. For perspective, the Hanford Site doses for 1992 may be compared to annual individual doses of 300 mrem and an annual collective dose of about 100,000 person-rem from natural background radiation.

5.1.2 Decentralization Alternative

The Decentralization Alternative would consider additional facility upgrades over those considered in the No Action Alternative, specifically, new wet storage (for defense production fuel only) or dry storage facilities, fuel stabilization via shear/leach/calcination or shear/leach/ solvent extraction, with research and development activities to support SNF management.

Impacts from storage prior to implementation of new wet or dry storage or fuels stabilization would not differ from those indicated for the No Action Alternative. In the event new storage facilities are selected some impacts would be associated with construction of those facilities. A proposed site has been identified comprising one-quarter section of land adjacent to the 200-East Area where any new facilities associated with SNF storage or stabilization that might be necessary would be assumed to be built. The area has been surveyed both for threatened and endangered species and for the presence of cultural resources; none were found. However, one federal candidate species, the loggerhead shrike, and one state candidate species, the sage sparrow, were seen. Use of this area is consistent with the Hanford mission and would impact no threatened or endangered biota. Construction would take place on up to 15 percent of the selected site. Construction activities would result in dust generation and various amounts of pollutants released from diesel-fueled equipment; however, concentrations at points of public access are expected to be well below permissible levels. Impacts associated with SNF storage would be expected to be less than those in the No Action Alternative.

Research and development of technologies for SNF stabilization would be undertaken in existing hot cell facilities in the 300 Area. Although not examined in detail for this programmatic analysis, no important environmental consequences have resulted from work in these facilities and none would be anticipated for development activities related to fuel processing.

5.1.3 1992/1993 Planning Basis Alternative

The 1992/1993 Planning Basis Alternative differs from the Decentralization Alternative only in that TRIGA fuel currently stored at the Hanford Site would be shipped to INEL for storage. The storage and stabilization options identified for the Decentralization Alternative are also assumed for the 1992/93 Planning Basis Alternative and that discussion is not repeated here. The potential impacts of transportation of TRIGA fuel to INEL are covered in Appendix I.

5.1.4 Regionalization Alternative

The Regionalization Alternative as it applies to the Hanford Site contains the following options:

A) All SNF, except defense production SNF, would be sent to INEL.

B1) All SNF west of the Mississippi River, except Naval SNF would be sent to Hanford.

B2) All SNF west of the Mississippi River and Naval SNF would be sent to Hanford.

C) All Hanford SNF would be sent to INEL or Nevada Test Site (NTS).

Facilities and features of Regionalization A would be the same as those described for Hanford defense production fuel in the Decentralization Alternative. The facilities and features for all other Hanford SNF would be very similar to those described for that spent nuclear fuel in the Centralization Minimum Alternative.

Facilities and features of Regionalization B1 and B2 options would be incremental to those described for the Decentralization Alternative and would be similar, but not identical, to those described in the Centralization Maximum Alternative.

Facilities and features of Regionalization C would be equivalent to those described for the Centralization Minimum Alternative.

5.1.5 Centralization Alternative

Two options exist at the Hanford Site for the Centralization Alternative: 1) shipment of all fuel within the DOE complex to the Hanford Site for management and storage, and 2) shipment of all fuel off of the Hanford Site. In the former option, dry storage of all fuel sent to the Hanford Site from offsite would be assumed. A facility equivalent to the decentralization suboptions would be assumed for processing of SNF prior to storage; fuel received from offsite would have been stabilized for dry storage prior to receipt. The consequences of implementing this option would be larger than those of the Decentralization Alternative. In the option of transferring all Hanford fuel to another site, a fuel stabilization and packaging facility would need to be constructed to prepare existing fuel for shipment.

5.2 Land Use

Implications of implementing the alternatives for interim storage of SNF on land use at the Hanford Site are discussed in the following subsections.

5.2.1 No Action Alternative

No new SNF facilities would be built at the Hanford Site; thus, land use patterns would remain as described in Section 4.2 and have no impact on the existing environment. The Hanford Site would remain a federal facility dedicated to nuclear research and development and environmental cleanup. Other continuing activities would include waste management, commercial power production, ecological research, and wildlife management, as described in Section 4.2.

5.2.2 Decentralization Alternative

This alternative would require the construction of an SNF facility for fuel management and storage. Most SNF from the Hanford Site would be stored at that facility.

Historically, the Hanford Site has been used for nuclear materials production. The construction and operation of an SNF facility would be consistent with this historical use. Off-site land use would not be affected by construction and operations of an SNF facility, except to the extent that some undeveloped lands probably would be developed for worker housing. Such development would be subject to local land use and zoning controls, which vary by jurisdiction. No project facilities would be located offsite.

No direct or indirect effects would occur to wildlife refuges on the Hanford Site because SNF activities would not be close to these areas. Similarly, no direct or indirect effects would occur to the Columbia River. Although construction at the SNF site would disturb native vegetation (Section 5.9.1), on up to 7 hectares (18 acres) of the 65-hectare (160-acre) site, this would involve only a small part of similar natural habitat at Hanford. The use of Hanford as a National Environmental Research Park would not be significantly affected.

No impacts requiring mitigation would occur to land uses a result of construction or operation of an SNF facility at the Hanford Site.

5.2.3 1992/1993 Planning Basis Alternative

The 1992/1993 Planning Basis Alternative differs from the Decentralization Alternative only in that TRIGA fuel currently stored at the Hanford Site may be shipped to INEL for storage. Thus, land use would be essentially the same as in the Decentralization Alternative. Although construction at the SNF site would disturb native vegetation (Section 5.9.1), on up to 7 hectares (18 acres) of the 65-hectare (160-acre) site, this would involve only a small part of similar natural habitat at Hanford. The use of Hanford as a National Environmental Research Park would not be significantly affected.

5.2.4 Regionalization Alternative

Construction of facilities in support of the Regionalization Alternative as it applies to the Hanford Site would result in the following disturbance of native vegetation and land use commitments:

A) From about 2 to 7 hectares (6 to 18 acres) when all SNF, except defense production SNF would be sent to INEL.

B1) From about 14 to 17 hectares (36 to 43 acres) when all SNF west of the Mississippi River, except Naval SNF would be sent to Hanford.

B2) From about 24 to 27 hectares (61 to 68 acres) when all SNF west of the Mississippi River and Naval SNF would be sent to Hanford.

C) From about 2 to 5 hectares (6 to 12 acres) when all Hanford SNF would be sent to INEL or NTS.

These areas involve only a small part of similar natural habitat at Hanford. The use of Hanford as a National Environmental Research Park would not be significantly affected.

5.2.5 Centralization Alternative

If Hanford is selected as the site for implementing the Centralization Alternative, the SNF facility and its support facilities (including a new Expended Core Facility) would be constructed. The impacts of such construction would be essentially the same as those presented for the Decentralization Alternative. Although construction at the SNF site would disturb native vegetation (Section 5.9.1) on up to 37 hectares (93 acres) of the 65-hectare (160-acre) site, this would involve only a small part of similar natural habitat at Hanford. In addition to the above total, new construction would also include construction of a new Expended Core Facility for fuel from the Naval Nuclear Propulsion Program. The use of Hanford as a National Environmental Research Park would not be significantly affected.

If Hanford is not selected as the site for centralization of SNF, an SNF stabilization and packaging facility would be built to prepare the fuel for transport offsite. This facility would have somewhat smaller construction requirements than would be required for storage of all DOE SNF at Hanford. The land use impacts would be similar to those described for the Regionalization option C.

5.2.6 Effects of Alternatives on Treaty or Other Reserved Rights of Indian Tribes and Individuals

The Yakama Indian Nation and the Confederated Tribes of the Umatilla Indian Reservation acquired certain rights and privileges in the 1855 treaty. These rights and privileges are also claimed by the Wanapum Tribe. In Article III, of the 1855 treaty it states that "The exclusive right of taking fish in all streams, where running through or bordering said reservation, is further secured to said confederated tribes and bands of Indians, as also the right of taking fish at all usual and accustomed places, in common with citizens of the Territory, and of erecting temporary buildings for curing them; together with the privilege of hunting, gathering roots and berries, and pasturing their horses and cattle upon open unclaimed land.[a]"

Although access to the Hanford Site has been restricted, tribal members have expressed an interest in renewing their use of these resources in accordance with the Treaty of 1855, and the DOE is assisting them in this effort. In keeping with this effort, each of the alternatives would provide for the rights and privileges identified in the treaty:

- Taking Fish - The alternatives considered in this document would not reduce access to fishing locations on the Hanford Site.
- Hunting, Gathering Roots and Berries, and Pasturing Livestock - The No Action Alternative would not further reduce the areas potentially available for hunting, gathering roots and berries, or pasturing livestock. All existing fenced areas assigned for SNF storage and a suitable buffer zone would likely remain unavailable for these activities. All other alternatives would require the construction of new facilities. This would further reduce the land base available for hunting, gathering, and pasturing. This impact could be on the order of 18 acres.

5.3 Socioeconomics

The following section describes the socioeconomic impacts of the SNF project at the Hanford Site. For the analysis, a ten-county region of influence was identified. While the region of influence covers the counties of Adams, Benton, Columbia, Franklin, Grant, Walla Walla, and Yakima in the state of Washington; and Morrow, Umatilla, and Wallowa counties in

a. These treaty rights and privileges are subject to diverse interpretations. None of the lands contemplated for use for SNF processing and/or storage at Hanford were on "open unclaimed land" when the government established the Hanford Site.

the state of Oregon, the majority of the impacts would be confined to the Benton-Franklin County region and the Tri-Cities (Richland, Kennewick, and Pasco) (see Figure 4-2).

The socioeconomic impacts are classified in terms of direct and secondary effects. Changes in Hanford employment and expenditures are classified as direct effects, while changes that result from Hanford regional purchases, nonpayroll expenditures, and payroll spending by Hanford employees are classified as secondary effects. The total socioeconomic impact within the region is the sum of the direct and secondary effects.

Estimates of total employment impacts were calculated using the Regional Input-Output Modeling System developed for the Hanford region of influence by the U.S. Bureau of Economic Analysis. This assessment reports the changes in employment and earnings based on historic data, which indicate that 93 percent of Hanford employees reside in the Benton-Franklin county area. Table 4.3-1 in Section 4.3 presents the baseline projections from which comparisons can be made.

All employment comparisons are made relative to the regional employment projections and not current Hanford Site employment projections. While a down-turn in Hanford Site employment is anticipated, the extent of the down-turn is unknown. The effect of such a down-turn on the region's employment projection used in this analysis is expected to be minimal because the regional projection, released in 1992, assumed a more stable rate of growth than the actual "boom" experienced in recent years.

5.3.1 No Action Alternative

Under the No Action Alternative, only the minimum actions required for continued safe and secure storage of SNF would occur. No new facilities would be constructed, and only minimal facility upgrades would take place. It is assumed that existing personnel would be utilized under this alternative, and therefore no incremental socioeconomic consequences are anticipated. Socioeconomic conditions would continue as described in Section 4.3.

5.3.2 Decentralization Alternative

Under the Decentralization Alternative, significant facility development and upgrades are permitted, with various suboptions defined for processing and storage of the SNF. The socioeconomic consequences related to implementing the decentralization alternatives are described in this subsection. The employment and population impacts related to construction and operation of the Decentralization Alternative suboptions are presented in Table 5.3-1. It was assumed that up to 300 current Hanford workers could be reassigned to operation activities (this number excludes current workers at the Fast Flux Test Facility because it was assumed that they would be reassigned to activities related to the Hanford Waste Vitrification Plant). Construction activities were assumed to require new workers coming into the area. Estimates of direct jobs were provided by Bergsman (1995). For construction activity, direct jobs were reported as number of jobs in the peak year and total person-years because it was assumed that construction activities would "ramp-up" to the peak year, and then "ramp-down," with the total number of jobs related to construction activity equaling the total person-years required, as reported in Bergsman (1995). Increases in activity levels could strain an already tight housing market and add to school-capacity concerns. However, because construction activities are short-term relative to the total project time frame, impacts from construction activities may be overstated.

5.3.2.1 Employment. All construction activity is assumed to peak in 1998. Construction activity for storage options W, X, Y, and Z occurs in the years 1997-2000; construction activity for processing suboptions P and Q occurs in the years 1998-2001. Increases in employment range from 221 (suboption X) to 1,094 (suboptions Y and P) and equate to between 0.3 and 1.3 percentage points over baseline regional employment projections (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity tapering off. Increases in employment range from 442 (suboptions Z and P) to 880 (suboptions Q and Small Vault) persons and equate to between 0.5 and 1.0 percentage points over baseline regional employment projections. Beyond 2004, operations activity will taper off as processing activities (suboptions P and Q) will occur only through 2005. Suboptions Y and Z each require only 50 workers beyond 2005 for operations activity. Because it is anticipated that up to 300 current workers could be reassigned, no incremental socioeconomic impacts are anticipated after 2005. This is also true with suboptions W and X because they are assumed to absorb between 200 and 210 current workers for the first two years of operation (2001-2002), with employment requirements falling to between 150 and 95

Table 5.3-1. Comparison of the socioeconomic impacts of spent nuclear fuel Decentralization Alternative suboptions.

Decentralization Alternative	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Suboption W										
Direct Jobs	0	0	216	251	216	181	0	0	0	0
Secondary Jobs	0	0	240	280	240	200	0	0	0	0
Population Change	0	0	590	680	590	490	0	0	0	0
Suboption X										
Direct Jobs	0	0	200	221	200	178	0	0	0	0
Secondary Jobs	0	0	220	240	220	200	0	0	0	0
Population Change	0	0	540	600	540	490	0	0	0	0
Suboptions Y and P										
Direct Jobs	0	0	318	1,094	1,033	971	715	464	464	464
Secondary Jobs	0	0	350	1,200	1,130	1,070	780	590	590	590
Population Change	0	0	870	2,980	2,810	2,650	1,950	1,370	1,370	1,370
Suboptions Q and Small Vault										
Direct Jobs	0	0	62	947	934	920	872	880	880	880
Secondary Jobs	0	0	70	1,040	1,020	1,010	960	1,120	1,120	1,120
Population Change	0	0	170	2,580	2,540	2,510	2,380	2,610	2,610	2,610
Suboptions Z and P										
Direct Jobs	0	0	213	935	926	920	715	442	442	442
Secondary Jobs	0	0	230	1,030	1,020	1,010	780	570	570	570
Population Change	0	0	580	2,550	2,530	2,510	1,950	1,310	1,310	1,310
Suboptions Q and Cask										
Direct Jobs	0	0	45	917	917	917	872	822	822	822
Secondary Jobs	0	0	50	1,010	1,010	1,010	960	1,050	1,050	1,050
Population Change	0	0	120	2,500	2,500	2,500	2,380	2,430	2,430	2,430

workers in 2003 and 2004. For the remaining years (2005-2035), suboptions W and X each would require only 60 workers for operation activities.

5.3.2.2 Population. For construction-related activities, the population is expected to peak in 1998, with increases in population ranging from 600 (suboption X) to 2,810 (suboptions Y and P) and equating to between 0.4 and 1.7 percentage points over baseline projections (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity tapering off through 2007. Increases in population range from 1,310 (suboptions Z and P) to 2,610 (suboptions Q and Small Vault) persons and equate to between 0.7 and 1.5 percentage points over baseline projections for 2002.

5.3.3 1992/1993 Planning Basis Alternative

This alternative defines those activities that were already scheduled at the various sites for the transportation, receipt, processing, and storage of SNF. Under this alternative, no new spent fuel would be sent to the Hanford Site, but the TRIGA fuel would be shipped offsite. The upgrades of existing storage facilities, as defined in the Decentralization alternative, were already planned, so the impacts of the 1992/1993 Planning Basis Alternative are essentially the same as outlined in Subsection 5.3.2. Because of the shipment of TRIGA fuel, an additional two workers per year would be required over 3 years of operation; however, it was assumed that current personnel would be reassigned to fill these jobs; therefore, the incremental impacts would be the same as those presented in Table 5.3-1.

5.3.4 Regionalization Alternative

Under this alternative, SNF would be redistributed to candidate sites based on similarity of SNF types or region within the country. There are four possible cases: regionalization of SNF by fuel type (Regionalization A); regionalization in which all SNF currently stored in the western United States, or to be generated in the western United States, except Naval SNF would be sent to and stored at the Hanford Site (Regionalization B1); regionalization in which all SNF currently stored in the western United States, or to be generated in the western United States, and all Naval fuel would be sent to and stored at the Hanford Site (Regionalization B2); and regionalization in which all SNF currently located in the western United States, or to be generated in the western United States, including all Hanford SNF, would be sent to and stored at another location (Regionalization C).

5.3.4.1 Regionalization A. In this case, all SNF currently located at Hanford, except defense production fuel, would be sent to INEL. For the Hanford Site, the facility requirements for the N reactor and single-pass reactor fuel would be the same as those described in the Decentralization Alternative. Facilities for all other Hanford Site fuel would be similar to those described within the Centralization minimum alternative. The population and employment impacts related to Regionalization A are presented in Table 5.3-2.

5.3.4.1.1 Employment. All construction activity is assumed to peak in 1998. Construction activity for suboptions RAX, RAY, and RAZ occurs in the years 1997-2000 and construction activity for suboption P occurs in the years 1998-2001. Increases in employment range from 176 (suboption RAX) to 1,065 (suboption RAY and P) and equate to between 0.2 and 1.3 percentage points over baseline projections of regional employment (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity tapering off. Increases in employment range from 208 (suboption RAY and P) to 230 (suboption RAZ and P) persons and equate to between 0.2 and 0.3 percentage points over baseline projections. Beyond 2004, operations activity will taper off as processing activities (suboption P) will only occur through 2005. Suboptions RAY and RAZ each require only 50 workers beyond 2005 for operations activity. Because it is anticipated that up to 300 current workers could be reassigned, no incremental socioeconomic impacts are anticipated after 2005. This is also true with suboption RAX because it would require only 59 workers for operation activities after 2005.

5.3.4.1.2 Population. For construction-related activities, the population is expected to peak in 1998, with increases in population ranging from 480 (suboption RAX) to 2,900 (suboption RAY and P) and equating to between 0.3 and 1.7 percentage points over baseline projections (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity tapering off through 2006. Increases in population range from 620 (suboption RAX) to 680 (suboption RAY and P) persons and equate to between 0.3 and 0.4 percentage points over baseline projections for 2002.

Table 5.3-2. Comparison of socioeconomic impacts of spent nuclear fuel Regionalization A suboptions.

Regionalization A Suboptions	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Suboption RAX										
Direct Jobs	0	0	90	176	176	176	0	0	0	0
Secondary Jobs	0	0	100	190	190	190	0	0	0	0
Population Change	0	0	250	480	480	480	0	0	0	0
Suboption RAY and P										
Direct Jobs	0	0	150	1,065	1,065	1,065	715	208	208	208
Secondary Jobs	0	0	160	1,170	1,170	1,170	780	270	270	270
Population Change	0	0	410	2,900	2,900	2,900	1,950	620	620	620
Suboption RAZ and P										
Direct Jobs	0	0	150	865	865	865	715	230	230	230
Secondary Jobs	0	0	160	950	950	950	780	290	290	290
Population Change	0	0	410	2,360	2,360	2,360	1,950	680	680	680

5.3.4.2 Regionalization B1. In this case, all SNF currently stored or to be generated in the western United States, except Naval SNF, would be sent to and stored at the Hanford Site. Facility requirements for this case would be incremental to those described for the Decentralization Alternative. Additional facilities include a storage facility for offsite fuel, a receiving and canning facility, and a technology development facility (RB1). The population and employment impacts related to regionalization B1 are presented in Table 5.3-3.

5.3.4.2.1 Employment. All construction activity is assumed to peak in 2000. Construction activity for suboptions W, X, Y, and Z occurs in the years 1997-2000; construction activity for suboptions P and Q occurs in the years 1998-2001; and construction of the additional facilities (suboption RB1) for receiving and canning and technology development occurs in the years 1998-2001, with 90% of the storage facility being constructed during the years 2000-2010 and the remaining 10% being constructed during the years 2010-2035. Increases in employment range from 398 (suboption X and RB1) to 1,191 (suboption Y and P and RB1) and equate to between 0.5 and 1.4 percentage points over baseline projections of regional employment (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity tapering off. Increases in employment range from 73 (suboption X and RB1) to 1,050 (suboption Q and Small Vault and RB1) persons and equate to between 0.1 and 1.2 percentage points over baseline projections. Beyond 2004, operations activity will taper off as described in Section 5.3.2.2.1.

5.3.4.2.2 Population. For construction-related activities, the population is expected to peak in 2000, with increases in population ranging from 1,090 (suboptions W and RB1 and X and RB1) to 3,250 (suboption Y and P and RB1) and equating to between 0.6 and 1.9 percentage points over baseline projections (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity tapering off through 2006. Increases in population range from 200 (suboptions X and RB1) to 3,100 (suboptions Q, Small Vault, and RB1) persons and equate to between 0.1 and 1.7 percentage points over baseline projections for 2002.

5.3.4.3 Regionalization B2. In this case, all fuel currently stored or to be generated in the western United States, including Naval fuel, would be sent to and stored at the Hanford Site. Facility requirements for this case would be essentially the same as those described in the Regionalization B1 case, as the only difference would be the presence of Naval fuel. The receiving and canning facility, offsite storage facility, and technology development facility are referred to as suboption RB2. Also required for this case is the Naval Nuclear Propulsion

Table 5.3-3. Comparison of socioeconomic impacts of spent nuclear fuel Regionalization B1 suboptions.

Regionalization B1 Suboption	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Suboptions W and RB1										
Direct Jobs	0	0	216	381	352	401	215	75	72	72
Secondary Jobs	0	0	240	420	390	440	240	80	80	80
Population Change	0	0	590	1,040	960	1,090	590	210	200	200
Suboptions X and RB1										
Direct Jobs	0	0	200	351	336	398	215	73	72	72
Secondary Jobs	0	0	220	390	370	440	240	80	80	80
Population Change	0	0	540	960	910	1,090	590	200	200	200
Suboptions Y, P, and RB1										
Direct Jobs	0	0	318	1,224	1,169	1,191	930	637	636	636
Secondary Jobs	0	0	350	1,340	1,280	1,310	1,020	800	800	800
Population Change	0	0	870	3,340	3,180	3,250	2,530	1,870	1,870	1,870
Suboptions Z, P, and RB1										
Direct Jobs	0	0	213	1,065	1,064	1,140	930	615	614	614
Secondary Jobs	0	0	230	1,170	1,170	1,250	1,020	770	770	770
Population Change	0	0	580	2,900	2,900	3,110	2,530	1,800	1,800	1,800
Suboptions Q, Small Vault, and RB1										
Direct Jobs	0	0	62	1,077	1,070	1,140	1,090	1,050	1,050	1,050
Secondary Jobs	0	0	70	1,180	1,170	1,250	1,190	1,330	1,330	1,330
Population Change	0	0	170	2,940	2,920	3,110	2,960	3,100	3,100	3,100
Suboptions Q, Cask, and RB1										
Direct Jobs	0	0	45	1,047	1,053	1,137	1,087	995	994	994
Secondary Jobs	0	0	50	1,150	1,150	1,250	1,190	1,260	1,260	1,260
Population Change	0	0	120	2,850	2,870	3,100	2,960	2,930	2,930	2,930

Program's Expended Core Facility (ECF). Discussion on the relocation of the ECF to the Hanford Site is provided in Appendix D to the INEL Spent Nuclear Fuel PEIS and is not included here. Population and employment impacts of the Regionalization B2 case are presented in Table 5.3-4.

5.3.4.3.1 Employment. All construction activity is assumed to peak in 2000. Construction activity for suboptions W, X, Y, and Z occurs in the years 1997-2000; construction activity for suboptions P and Q occurs in the years 1998-2001; and construction of the additional facilities (suboption RB1) for receiving and canning and technology development occurs in the years 1998-2001, with 35% of the storage facility being constructed during the years 2000-2010 and the remaining 65% being constructed during the years 2010-2035. Increases in employment range from 488 (suboptions X and RB2) to 1,281 (suboptions Y, P, and RB2) and equate to between 0.6 and 1.5 percentage points over baseline projections of regional employment (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity tapering off. Increases in employment range from 80 (suboptions X and RB2) to 1,085 (suboptions Q, Small Vault, and RB2) persons and equate to between 0.1 and 1.3 percentage points over baseline projections. Beyond 2004, operations activity will taper off as described in section 5.3.2.2.1.

5.3.4.3.2 Population. For construction-related activities, the population is expected to peak in 2000, with increases in population ranging from 1,330 (suboptions X and RB2) to 3,490 (suboptions Y, P and RB2) and equating to between 0.8 and 2.0 percentage points over baseline projections (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity tapering off through 2006. Increases in population range from 220 (suboption X and RB2) to 3,190 (suboptions Q, Small Vault, RB2) persons and equate to between 0.1 and 1.8 percentage points over baseline projections for 2002.

5.3.4.4 Regionalization C. In this case, all fuel currently stored or to be generated in the western United States, including all Hanford Site fuel, would be sent to and stored at INEL or NTS. Facility requirements for the Hanford Site in this case are identical to those described in the Centralization Minimum Alternative. Employment and population impacts of this case are provided in Table 5.3-5 and are discussed in Section 5.3.5.2.

Table 5.3-4. Comparison of socioeconomic impacts of spent nuclear fuel Regionalization B2 suboptions.

Regionalization Alternative	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Suboptions W and RB2										
Direct Jobs	0	0	216	451	446	491	310	107	80	80
Secondary Jobs	0	0	240	490	490	540	340	120	90	90
Population Change	0	0	590	1,230	1,220	1,340	850	300	220	220
Suboptions X and RB2										
Direct Jobs	0	0	200	421	430	488	310	80	80	80
Secondary Jobs	0	0	220	460	470	540	340	90	90	90
Population Change	0	0	540	1,150	1,170	1,330	850	220	220	220
Suboptions Y, P, and RB2										
Direct Jobs	0	0	318	1,294	1,263	1,281	1,025	669	669	669
Secondary Jobs	0	0	350	1,420	1,380	1,400	1,120	840	840	840
Population Change	0	0	870	3,530	3,440	3,490	2,790	1,960	1,960	1,960
Suboptions Z, P, and RB2										
Direct Jobs	0	0	213	1,135	1,158	1,230	1,025	647	647	647
Secondary Jobs	0	0	230	1,240	1,270	1,350	1,120	810	810	810
Population Change	0	0	580	3,090	3,150	3,350	2,790	1,900	1,900	1,900
Suboptions Q, Small Vault and RB2										
Direct Jobs	0	0	62	1,147	1,164	1,230	1,182	1,085	1,085	1,085
Secondary Jobs	0	0	70	1,260	1,280	1,350	1,300	1,370	1,370	1,370
Population Change	0	0	170	3,130	3,170	3,350	3,220	3,190	3,190	3,190
Suboptions Q, Cask, and RB2										
Direct Jobs	0	0	45	1,117	1,147	1,227	1,182	1,027	1,027	1,027
Secondary Jobs	0	0	50	1,230	1,260	1,350	1,300	1,300	1,300	1,300
Population Change	0	0	120	3,040	3,130	3,340	3,220	3,020	3,020	3,020

Table 5.3-5. Comparison of socioeconomic impacts of spent nuclear fuel Centralization Alternative - maximum case suboptions.

Centralization Alternative	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Suboptions W and CM										
Direct Jobs	0	0	216	626	606	611	430	242	193	193
Secondary Jobs	0	0	240	690	660	670	470	280	220	220
Population Change	0	0	590	1,710	1,650	1,670	1,170	680	540	540
Suboptions X and CM										
Direct Jobs	0	0	200	596	590	608	430	164	135	135
Secondary Jobs	0	0	220	650	650	670	470	180	150	150
Population Change	0	0	540	1,620	1,610	1,660	1,170	450	360	360
Suboptions, Y, P, and CM										
Direct Jobs	0	0	318	1,469	1,423	1,401	1,145	804	804	804
Secondary Jobs	0	0	350	1,610	1,560	1,540	1,260	1,000	1,000	1,000
Population Change	0	0	870	4,000	3,880	3,820	3,120	2,350	2,350	2,350
Suboptions Z, P, and CM										
Direct Jobs	0	0	213	1,310	1,318	1,350	1,145	782	782	782
Secondary Jobs	0	0	230	1,440	1,440	1,480	1,260	970	970	970
Population Change	0	0	580	3,570	3,590	3,680	3,120	2,280	2,280	2,280
Suboptions Q, Small Vault,										
Direct Jobs	0	0	62	1,322	1,324	1,350	1,302	1,220	1,220	1,220
Secondary Jobs	0	0	70	1,450	1,450	1,480	1,430	1,530	1,530	1,530
Population Change	0	0	170	3,600	3,610	3,680	3,550	3,580	3,580	3,580
Suboptions Q, Cask, and CM										
Direct Jobs	0	0	45	1,292	1,307	1,347	1,302	1,162	1,162	1,162
Secondary Jobs	0	0	50	1,420	1,430	1,480	1,430	1,460	1,460	1,460
Population Change	0	0	120	3,520	3,560	3,670	3,550	3,410	3,410	3,410

5.3.5 Centralization Alternative

Under this alternative, all current and future SNF would be stored at a centralized location. There are two possible options: the maximum option in which all fuel is stored at Hanford, and the minimum option in which all fuel at Hanford is shipped offsite. The socio-economic consequences related to implementing the Centralization Alternative suboptions are described in this subsection. The employment and population impacts related to construction and operation of the maximum option are presented in Table 5.3-5. The population and employment impacts related to construction and operation of the minimum option are presented in Table 5.3-6. It was assumed that up to 300 current Hanford workers could be reassigned to operation activities (this number excludes current workers at the Fast Flux Test Facility, as it was assumed that they would be reassigned to activities related to the Hanford Waste Vitrification Plant). Construction activities were assumed to require new workers coming into the area. Estimates of direct jobs were provided by Bergsman (1995). For construction activity, direct jobs were reported as number of jobs in the peak year and total person-years because it was assumed that construction activities would "ramp-up" to the peak year, and then "ramp-down," with the total number of jobs related to construction activity equaling the total person-years required as reported in Bergsman (1995). Although the housing market is currently uncertain and beginning to turn downward, increases in activity levels could strain the housing market and add to school-capacity concerns. However, because construction activities are short-term relative to the total project time frame, impacts from construction activities may be overstated.

5.3.5.1 Centralization - Maximum Option. Under the maximum option, Hanford SNF would be stabilized and stored under one of the options outlined in the decentralization alternative, with larger storage facilities. A facility would also be built to receive SNF from other sites. Additionally, the ECF would be relocated from the INEL site. The impacts of the ECF to regional population and employment are presented in Appendix D of Volume 1 of this EIS and are not discussed here. Table 5.3-5 presents the employment and population impacts of the options under the maximum centralization option.

5.3.5.1.1 Employment. All construction activity is assumed to peak in 2000. Construction activity for suboptions W, X, Y, and Z occurs in the years 1997-2000; construction activity for suboptions P and Q occurs in the years 1998-2001; and construction activity for the

Table 5.3-6. Comparison of socioeconomic impacts of spent nuclear fuel Centralization Alternative - minimum case suboptions.

Centralization Alternative	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Suboption P										
Direct Jobs	0	0	0	715	715	715	715	360	360	360
Secondary Jobs	0	0	0	780	780	780	780	460	460	460
Population Change	0	0	0	1,950	1,950	1,950	1,950	1,070	1,070	1,070
Suboption Q										
Direct Jobs	0	0	0	872	872	872	872	786	786	786
Secondary Jobs	0	0	0	960	960	960	960	1,000	1,000	1,000
Population Change	0	0	0	2,380	2,380	2,380	2,380	2,330	2,330	2,330
Suboption D										
Direct Jobs	0	0	619	620	619	619	357	357	357	357
Secondary Jobs	0	0	680	680	680	680	460	460	460	460
Population Change	0	0	1,690	1,690	1,690	1,690	1,060	1,060	1,060	1,060

receiving and canning facility (suboption CM) occurs in the years 1998-2001, with 50% of the construction activity for the modular storage facility occurring during the years 2000-2010 and the other 50% occurring during the years 2010-2035. Increases in employment range from 608 (suboptions X and CM) to 1,401 (suboptions Y, P, and CM) and equate to between 0.7 and 1.7 percentage points over baseline projections of regional employment (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity tapering off. Increases in employment range from 164 (suboptions X and CM) to 1,220 (suboptions Q, Small Vault, and CM) persons and equate to between 0.2 and 1.4 percentage points over baseline projections. Beyond 2004, operations activity will taper off as processing activities (suboptions P and Q) will occur only through 2005. Operation of the receiving and canning facility will require 190 workers through 2011, falling to 150 workers through 2035. Suboptions Y and Z each require only 50 workers beyond 2005 for operations activity. Because it is anticipated that up to 300 current workers could be reassigned, no incremental socioeconomic impacts are anticipated after 2005. This is also true with suboptions W and X because each would require only 60 workers for operation activities.

5.3.5.1.2 Population. For construction-related activities, the population is expected to peak in 2000, with increases in population ranging from 1,620 (suboptions X and CM) to 3,818 (suboptions Y, P, and CM) and equating to between 0.9 and 2.2 percentage points over baseline projections (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity tapering off through 2007. Increases in population range from 450 (suboptions X and CM) to 3,580 (suboptions Q, Small Vault, and CM) persons and equate to between 0.3 and 2.0 percentage points over baseline projections for 2002.

5.3.5.2 Centralization. Minimum Option. Under the minimum option, Hanford's SNF would be shipped offsite. Some stabilization of fuel would be required prior to shipment of N Reactor and single-pass reactor fuel. Three options were identified for the stabilization: a shear/leach/calcine facility (suboption P); a solvent extraction facility (suboption Q); or a drying and passivation facility (suboption D). Suboptions P and Q are the same processing facilities that were included in the Decentralization Alternative. Table 5.3-6 presents the employment and population impacts of the suboptions under the Centralization minimum option.

5.3.5.2.1 Employment. All construction activity is assumed to peak in 1998. Construction activity for suboptions P and Q occurs in the years 1998-2001. Increases in employment range from 620 (suboption D) to 872 (suboption Q) and equate to between 0.7 and

1.0 percentage points over baseline projections (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity ending after 2006 for suboptions P and Q, and after 2004 for suboption D. Increases in employment range from 357 (suboption D) to 786 (suboption Q) persons and equate to between 0.4 and 0.9 percentage points over baseline projections.

5.3.5.2.2 Population. For construction-related activities, the population is expected to peak in 1998, with increases in population ranging from 1,690 (suboption D) to 2,380 (suboption Q) and equating to between 1.0 and 1.4 percentage points over baseline projections (see Table 4.3-1). All operations activity peaks in 2002, with incremental activity ending after 2006. Increases in population range from 1,060 (suboption D) to 2,330 (suboption Q) persons and equate to between 0.6 and 1.3 percentage points over baseline projections for 2002.

5.4 Cultural Resources

The potential impacts of SNF management activities on cultural resources were assessed by 1) identifying project activities that could directly or indirectly affect significant resources; 2) identifying the known or expected significant resources in areas of potential impact; and 3) determining whether a project activity would have no effect, no adverse effect, or an adverse effect on significant resources (36 CFR 800.9). Direct impacts are considered to be those associated with ground disturbance or activities that would destroy or modify an architectural structure. Indirect impacts are considered to be those resulting from improved visitor access, changes in land status, or other actions that limit scientific investigation of the resources.

Possible measures that would be worked out in consultation with the Washington State Historic Preservation Officer (SHPO), Advisory Council for Historic Preservation, and area tribes may include avoidance or data recovery.

5.4.1 No Action Alternative

The No Action Alternative would not involve upgrade or expansion of existing facilities, other than those that may be required to ensure safety and security. Specific actions considered in the No Action Alternative include continued storage at the following facilities:

- 105-KE and 105-KW Basins
- T Plant
- FFTF
- 308 Building
- 324 Building
- 325 Building
- 327 Building
- Low-Level Burial Grounds.

With the exception of FFTF, these are existing Manhattan Project and/or Cold War facilities currently under evaluation for National Register of Historic Places (NRHP) eligibility.

No new facilities would be required; however, the following facility modifications would be considered:

- Upgrade water supply and distribution system to 100-K Area.
- Upgrade seismic adequacy of K Basins.
- Upgrade fire protection systems for the K Basins.
- Safeguards and security upgrades to the K Basins.

Upgrade of the water supply and distribution system has the potential to adversely affect prehistoric archaeological sites in the vicinity of the 100-K Area. Several archaeological sites (45BN115, 45BN152, 45BN423, 45BN434, 45BN464, 45BN424, and H3-10) have been identified in this area (Chatters et al. 1992). These sites are being evaluated for their National Register eligibility. A careful review of the detailed project plans is necessary prior to initiation of this work. If the upgrade results in ground disturbance, as in the replacement and/or addition of new water lines, then these actions could directly affect the archaeological sites. However, proper design of the upgrade system could allow for avoidance of these prehistoric sites. If avoidance is not possible, some sort of data recovery or other measures may be developed in conjunction with affected Native American Tribes and the SHPO. The remaining facility modifications are not likely to affect the historical or architectural value of the Manhattan Project and/or Cold War facilities.

Some indirect effects might result from the continued operation of SNF storage facilities by Hanford workers in the culturally sensitive 100-K Area, if unauthorized artifact collection would contribute to the degradation of nearby archaeological sites. These effects could be mitigated through a worker education program, which would use posters to inform workers of applicable laws, briefing sessions for all persons expected to work along the corridor, and penalties for disturbing an archaeological site. The briefing sessions would stress the importance of cultural resources and specifics of the laws and regulations that exist for site protection.

Direct or indirect impacts are not anticipated to any known traditional cultural resources that are significant to members of the Yakama Indian Nation, the Confederated Tribes of the Umatilla Indian Reservation, or the Wanapum Band. This conclusion is based on the proposed locations of facilities relative to sacred and culturally important areas identified through ethno-historical research and interviews with elders of bands that formerly used the Hanford Site (Chatters 1989).

5.4.2 Decentralization Alternative

This alternative would involve additional facility upgrades beyond those described for the No Action Alternative, including the construction of new storage facilities and/or a processing facility. Several suboptions have been proposed that would require construction of new facilities. Table 5.4-1 lists the various suboptions and their facility requirements.

Table 5.4-1. Facility requirements of Decentralization suboptions and estimations of area disturbed, [hectares (acres)].

Sub-options	Process option	New pool	New dry vault	New dry casks	New process facility	New land disturbed
W	None	2.4 (6)	2.4 (6)			4.9 (12)
X	None	2.4 (6)		2 (5)		4.5 (11)
Y	P		4.9 (12)		2.4 (6)	7.3 (18)
	Q		2.4 (6)		4.9 (12)	7.3 (18)
	D		4.9 (12)		2.4 (6)	7.3 (18)
Z	P			4.9 (12)	2.4 (6)	7.3 (18)
	Q			2 (5)	4.9 (12)	6.9 (17)
	D			4.9 (12)	2.4 (6)	7.3 (18)

All suboptions would require the temporary use of 105-KE and 105-KW basins for packaging of fuel prior to relocation to a new wet storage facility, or stabilization for dry storage. These are existing Manhattan Project and/or Cold War facilities (currently under evaluation for National Register eligibility). Modifications to these existing facilities are considered to be comparable to those identified in the No Action Alternative.

Actions during the upgrade of the water supply and distribution system for the 100-K Area that disturb ground have the potential to adversely affect prehistoric archaeological sites in the vicinity of the 100-K Area (45BN115, 45BN152, 45BN423, 45BN434, 45BN464, 45BN424, and H3-10). A review of specific upgrade actions is required to determine these effects prior to initiation of these actions. Design of the upgrade system should incorporate avoidance of these prehistoric sites. If avoidance is not possible, some sort of data recovery or other measures may be developed in conjunction with affected Native American Tribes, the SHPO, and the Advisory Council.

An indirect effect of continued operation and maintenance of these facilities is the potential for Hanford workers to conduct unauthorized artifact collection activities. This effect could be mitigated through a worker education program, which would use posters to inform workers of applicable laws, briefing sessions for all persons expected to work along the corridor, and penalties for disturbing an archaeological site. The briefing sessions would stress the importance of cultural resources and specifics of the laws and regulations that exist for site protection.

All of the suboptions would require the construction of new facilities. Wet storage pool and dry storage vault facilities would be cast-in-place concrete structures. The dry cask storage facility would consist of modular storage casks on a concrete pad. The stabilization facilities would be multilevel steel-reinforced, cast-in-place concrete structures. The total land area disturbed by the construction of these facilities is estimated to range from 11 to 18 acres.

All new facilities would be located on a 160-acre site just west of 200-East Area (Figure 4-1). The construction of these facilities is not expected to directly affect any archaeological resources. The proposed project area has been surveyed for cultural resources (HCRC 94-600-001), and no prehistoric or historic archaeological properties were found. Consultation with the State Historic Preservation Office and affected Native American Tribes is still in progress. No indirect effects would be anticipated either because no archaeological sites are known to occur within approximately 4 kilometers of the location proposed for the SNF storage facilities. The SNF facilities would be constructed in an industrialized area and would not alter the feeling or association of the Manhattan Project and/or Cold War facilities located nearby.

Text describing impacts to areas of known traditional or religious significance to specific Native American Tribes for the No Action Alternative in Subsection 5.4.1 also applies to the Decentralization Alternative.

5.4.3 1992/1993 Planning Basis Alternative

This alternative involves continued SNF onsite transportation, receipt, processing, and storage at the Hanford Site. However, the TRIGA fuel currently stored at Hanford would be shipped to INEL. The impacts to cultural resources caused by storage of this fuel at INEL are covered in Volume 1, Appendix B (INEL Spent Nuclear Fuel Management Program). The storage and stabilization facility options for Hanford under this alternative are assumed to be consistent with those of the Decentralization Alternative. Refer to Subsection 5.4.2 for a discussion of the cultural resource impacts.

5.4.4 Regionalization Alternative

All new facilities would be constructed on the 65 hectare (163-acre) site west of 200-East Area (Figure 4.1). Construction of these facilities is not expected to have a direct effect on any significant archaeologic resources. The proposed project area has been surveyed for cultural resources (HCRC 94-600-017), and no prehistoric or historic archaeological properties were found. Two isolated artifacts, one historic and one prehistoric in origin, were recorded during the inventory. Because of their isolated status, neither of the artifacts is considered significant. No indirect effects are anticipated because no known archaeological sites are present within approximately 4 kilometers (2 1/2 miles) of the location proposed for the SNF storage facilities.

Because the site for the new SNF facilities is in an industrialized area, construction of these facilities would not alter the feeling or association of the Manhattan Project and/or Cold War facilities located nearby.

Although no cultural resource impacts are expected, the potential for discovery during construction is proportional to the amount of land that would be disturbed. For the various options of the Regionalization Alternative, those areas would amount to the following amounts of land:

- A) From about 2 to 7 hectares (6 to 18 acres) when all SNF, except defense production SNF, would be sent to INEL
- B1) From about 14 to 17 hectares (36 to 43 acres) when all SNF west of the Mississippi River, with the exception of Naval SNF, would be sent to Hanford
- B2) From about 24 to 27 hectares (61 to 68 acres) when all SNF west of the Mississippi River and Naval SNF would be sent to Hanford
- C) About 2 to 5 hectares (6 to 12 acres) when all Hanford SNF would be sent to INEL or NTS.

In any event, the maximum option would require a processing facility (equivalent to Decentralization process options P, Q, or D) with a specialty fuel processing area; an inspection and packaging facility; an SNF storage complex (similar to, but larger than that for the Decentralization options W, X, Y, or Z); and a new Expended Core Facility. The existing 105-KE and 105-KW basins would be used to package fuel for wet transport to the processing facility. These are existing Manhattan Project and/or Cold War facilities that are currently under evaluation for National Register eligibility. Modifications to these facilities are considered to be similar to those depicted for the No Action and Decentralization alternatives (refer to Subsections 5.4.1 and 5.4.2). Ground-disturbing upgrades to the 100-K Area water supply and distribution system are considered to have potentially adverse effects on prehistoric archaeological sites 45BN115, 45BN152, 45BN423, 45BN434, 45BN424, H3-10, and/or 45BN464 located in this vicinity. A review of the specific upgrade plans is required to determine the effects before beginning these activities. Design of the upgraded water supply system should incorporate avoidance of the prehistoric sites. If avoidance is not possible, then some data recovery or other measures would be developed in conjunction with the affected Native

American Tribes, the SHPO, and the Advisory Council. Text describing potential unauthorized artifact collection and possible mitigation measures for the Decentralization Alternative in Subsection 5.4.2 also applies to the Regionalization Alternative.

Text describing impacts to areas of known traditional or religious significance to specific Native American Tribes for the No Action Alternative in Subsection 5.4.1 also applies to the Regionalization Alternative.

5.4.5 Centralization Alternative

This alternative consists of two scenarios: shipment of all SNF off of the Hanford Site (minimum option), and storage of all SNF at the Hanford Site (maximum option). For the minimum option, a new fuel stabilization and packaging (canning) facility would be constructed.

The maximum option would require a processing facility (equivalent to Decentralization process options P, Q, or D) with a specialty fuel processing area; an inspection and packaging facility; an SNF storage complex (similar to the decentralization options W, X, Y, or Z); and a new Expended Core Facility. The existing 105-KE and 105-KW Basins would be used to package defense production fuel for wet transport to the processing facility. These are existing Manhattan Project and/or Cold War facilities that are currently under evaluation for National Register eligibility. Modifications to these facilities are considered to be similar to those depicted for the No Action and Decentralization Alternatives (refer to Subsections 5.4.1 and 5.4.2). Ground-disturbing upgrades to the 100-K Area water supply and distribution system are considered to have potentially adverse effects on prehistoric archaeological sites 45BN115, 45BN152, 45BN423, 45BN434, 45BN424, H3-10, and/or 45BN464 located in this vicinity. A review of the specific upgrade plans is required to determine the effects before beginning these activities. Design of the upgraded water supply system should incorporate avoidance of the prehistoric sites. If avoidance is not possible, then some data recovery or other measures would be developed in conjunction with the affected Native American Tribes, the SHPO, and the Advisory Council. Text describing potential unauthorized artifact collection and possible mitigation measures for the Decentralization Alternative in Subsection 5.4.2 also applies to the Centralization Alternative.

All new facilities would be constructed on the 160-acre site west of 200-East Area (Figure 4.1). The construction of these facilities is not expected to have a direct effect on any archaeologic resources. The proposed project area has been surveyed for cultural resources (HCRC 94-600-001), and no prehistoric or historic archaeological properties were found. No indirect effects are anticipated because no known archaeological sites are present within approximately 4 kilometers of the location proposed for the SNF storage facilities. The site for the new SNF facilities is in an industrialized area, thus construction of these facilities would not alter the feeling or association of the Manhattan Project and/or Cold War facilities located nearby.

Text describing impacts to areas of known traditional or religious significance to specific Native American Tribes for the No Action Alternative in Subsection 5.4.1 also applies to the Centralization Alternative.

5.5 Aesthetic and Scenic Resources

Implications of implementing the alternatives for interim storage of SNF on aesthetic and scenic resources at the Hanford Site are discussed in the following subsections.

5.5.1 No Action Alternative

Impacts from this alternative would have no effect on the aesthetic and scenic resources.

5.5.2 Decentralization Alternative

This alternative would require the construction of an SNF facility at Hanford, where most SNF from the Hanford Site would be stored.

Changes caused by construction and operation of an SNF facility would be consistent with the existing overall visual environment of the Hanford Site. Topographic features obstruct the SNF site from view from populated areas. The site could be seen from the farmland bluffs that overlook the Columbia River on the east. However, these lands are on private property not readily accessible to the public. Landowners would likely grant access permission only during the hunting season, if at all.

No impacts requiring mitigation would occur to the aesthetics or to the visual environment as a result of construction or operation of an SNF facility at the Hanford Site.

5.5.3 1992/1993 Planning Basis Alternative

Activities in this alternative are sufficiently similar to those of the Decentralization Alternative that they are not repeated here.

5.5.4 Regionalization Alternative

This alternative (see Section 5.1.4 for details) would require the construction of a variety of SNF facilities depending on the option chosen. The facilities would range from a packaging/stabilization facility if all fuel were to be removed from Hanford (option C) to storage facilities for all SNF west of the Mississippi River (option B2). However, changes caused by construction and operation of these facilities would be consistent with the existing overall visual environment of the Hanford Site. Topographic features obstruct the SNF site from view from populated areas. The site could be seen from the farmland bluffs to the east of the site that overlook the Columbia River. However, these lands are on private property that is not readily accessible to the public. Landowners would likely grant access permission only during the hunting season, if at all.

No impacts requiring mitigation would occur to the aesthetics or to the visual environment as a result of construction or operation of an SNF facility at the Hanford Site.

5.5.5 Centralization Alternative

If Hanford is selected as the site for centralization of SNF, then the SNF facility and its support facilities would be constructed here.

Changes caused by construction and operation of an SNF facility would be substantially larger in the Centralization Maximum Alternative. However, they would be consistent with the existing overall visual environment of the Hanford Site. Topographic features obstruct the SNF site from view from populated areas. The site could be seen from the farmland bluffs that overlook the Columbia River on the east. However, these lands are on private property not readily accessible to the public. Landowners would likely grant access permission only during the hunting season, if at all.

No impacts requiring mitigation would occur to the aesthetics or to the visual environment as a result of construction or operation of an SNF facility at the Hanford Site. If Hanford is not selected as the site for centralization of SNF, only an SNF packaging/processing facility for shipment of fuel would be constructed and there would be even less potential for impact to the aesthetic and scenic resources.

5.6 Geologic Resources

No postulated impacts to the geologic resources of the Hanford Site have been identified under any of the alternatives. Thus, geologic resources would remain as described under Section 4.6.

5.7 Air Quality and Related Consequences

The consequences of the five alternatives on ambient air quality at the Hanford Site are presented in this section. In the case of radiological emissions, the consequences are compared among the alternatives and to current Hanford Site operations. For nonradiological emissions, projected ambient concentration at key receptor locations are compared with current concentrations at the Hanford Site. Development of the specific analysis for each alternative is discussed in subsequent subsections.

The consequences of radiological emissions were evaluated using the GENII computer code package (Napier et al. 1988). The radiological consequences of airborne emissions during normal operation have been estimated for the SNF storage alternatives considered in this document. Three separate analyses were performed for each facility included in a particular alternative using the GENII computer code. The receptors evaluated in these cases were at the location of maximum exposure representing a potential onsite worker outside of the SNF facility, the maximally exposed offsite resident, and the collective population within 80 kilometers. Standard parameters for radiological dose calculations at the Hanford Site were used for these estimates (Schreckhise et al. 1993). The maximum impact of each alternative on offsite receptors and workers was obtained by summing the consequences associated with the individual facilities, although these receptors may be physically at very different locations. The health consequences in terms of cancer fatalities were calculated using recommendations of the International Commission on Radiological Protection in its Publication 60 (ICRP 1991) - 4E-04 fatal cancers/rem for workers and 5E-04 fatal cancers/rem for the general population. Risk conversion factors were applied to both individual and collective doses, although they are based on population averages for individuals with varying degrees of sensitivity. The individual risk estimates therefore represent the risk to a hypothetical individual, which would be somewhat lower than the risk to more sensitive members of the population.

None of the alternatives would result in a dose to the maximally exposed offsite resident that exceeds 1 percent of the current EPA standard of 10 millirem/year. The consequences of the No Action Alternative are caused by emissions from existing facilities where spent fuel is stored. These facilities contribute a relatively small fraction of the total dose from airborne emissions at all Hanford Site operations (less than half and likely much less). The No Action Alternative represents the baseline for SNF operations at Hanford. The consequences of the Decentralization, Regionalization, and Centralization Alternatives vary depending on which storage and processing options are considered. Options including processing of defense reactor fuel result in the highest doses, which are at most an order of magnitude greater than those in the No Action Alternative. The consequences of options involving only containerization of defense reactor fuel followed by wet storage, and dry storage of all other fuel, in a new facility are approximately an order of magnitude lower than those in the No Action Alternative.

The potential nonradiological air quality pollutants of concern for this assessment include all pollutants for which there exist federal, state, or local standards. This includes both the standard set of criteria pollutants (e.g., nitrogen dioxide, oxides of sulfur, respirable particles) and toxic pollutants.

For criteria pollutants, concentration levels are regulated by the provisions of the Clean Air Act; Washington State standards for these criteria pollutants are at least as stringent as the federal standards. In the State of Washington, the Department of Ecology has the responsibility for promulgating and enforcing air quality standards for the protection of public health. The regulation that governs the control of toxic air pollutants (WAC 1990a,b) requires the owners of new or modified air emission sources to apply for approval before construction. Owners of sources emitting toxic air pollutants must demonstrate that they will employ the best available control technology for emissions control with reasonable environmental, energy, and economic impacts.

Construction of new facilities can also negatively impact air quality through the emission of fugitive dusts. To model this aspect, the EPA's Fugitive Dust Model (FDM) was selected. This model is especially designed to compute the air quality impacts from fugitive dust emissions, such as those associated with facility construction sites (Winges 1992). The FDM uses steady-state Gaussian plume algorithms and a gradient-transfer deposition algorithm to compute air quality impacts. Emissions for each source must be apportioned into a series of particle-size classes; each of which is assigned a representative deposition velocity. The model can operate using either joint frequency distributions or hourly meteorological data to represent atmospheric conditions. The model can handle up to 200 sources and 500 receptors per model run. The user may define a variety of point, line, area, and volume sources.

The Industrial Source Complex (ISC2) models were selected to estimate routine non-radiological air quality impacts. There are two ISC2 models: the ISC2 short-term model (ISCST2) and the ISC2 long-term model (ISCLT2). The two ISC2 models use steady-state Gaussian plume algorithms to estimate pollutant concentrations from a wide variety of sources associated with industrial complexes (EPA 1992). The models are appropriate for flat or rolling terrain, modeling domains with a radius of less than 50 kilometers, and urban or rural environments. The ISC2 models have been approved by the EPA for specific regulatory applications and are designed for use on personal computers. Input requirements for the ISC2 model include a variety of information that defines the source configuration and pollutant emission parameters. The user may define a variety of point, line, area, and volume sources. The ISCST2 model uses hourly meteorological data and joint frequency distribution data to compute straightline plume transport. Plume rise, stack-tip downwash, and building wake can be computed. The ISC2 models compute a variety of short- and long-term averaged products at user-specified receptor locations and receptor rings. The ISC2 models also treat deposition processes and allow the exponential decay of pollutants.

5.7.1 No Action Alternative

Facilities included in the No Action Alternative consist of those where SNF is currently stored at the Hanford Site. Minimal repackaging, stabilization, and relocation of fuel would be undertaken to ensure continued safe storage prior to ultimate disposition. The majority of spent fuel at Hanford is located at the 100-K Area wet storage basins. In addition, smaller quantities of fuel are stored at other onsite facilities. These include T Plant and a low-level waste burial ground in the 200-West Area; the Fast Flux Test Facility in the 400 Area; and the 308, 324, 325, and 327 buildings in the 300 Area. Releases for the No Action Alternative are based on operations for these facilities during 1992 (Bergsman 1995). These emissions were assumed to represent operations at existing SNF storage facilities over the EIS evaluation period, although they are subject to change with individual facility missions and operating status. It should also be noted that some existing facilities support a variety of other programs in energy research and waste management in addition to laboratory and hot cell examination of fuel materials. The historical releases from these multi-purpose facilities may reflect other activities in addition to spent fuel storage. The past operating emissions, therefore, represent an upper bound estimate for the fuel storage activities. The No Action Alternative also represents the baseline of maximum expected impacts for future spent fuel storage activities.

5.7.1.1 Radiological. Radiological air emissions for normal operation of existing fuel storage facilities in the No Action Alternative are listed in Tables 5.7-1 through 5.7-3 (DOE/RL 1993). The sealed fuel canisters temporarily stored at the 200-West Area burial ground are assumed to release negligible quantities of radionuclides in this analysis, although actual emissions from the stored fuel have not been quantified.

The consequences of air emissions from existing facilities utilized in the No Action Alternative are summarized in Table 5.7-4 and include a maximum annual dose of 1E-5 rem to a potential onsite worker with a 5E-9 probability of fatal cancer. The maximum dose to an offsite resident is estimated as 3E-6 rem/year, and the corresponding probability of fatal cancer is 1E-9. The dose estimate for an onsite worker or an offsite individual represents the sum of doses to separate maximally exposed individuals for each of the facilities included in the alternative. Because these facilities are in different areas of the Hanford Site, the respective maximally exposed workers and offsite residents are at different locations. The actual dose to a single worker or

Table 5.7-1. Annual atmospheric releases for normal operation - wet storage basins at 100-KE Area and 100-KW Area.

Radionuclide	100-KE Area Release (Ci/yr)	100-KW Area Release (Ci/yr)
Cobalt-60	1.3E-06	1.4E-06
Strontium-90	1.6E-04	9.9E-07
Ruthenium-106	1.3E-05	6.2E-06
Antimony-125	1.1E-05	NA[a]
Cesium-137	2.3E-04	2.7E-05
Europium-154	NA	4.9E-06
Plutonium-238	1.3E-06	3.0E-08
Plutonium-241	3.9E-05	NA
Americium-241	5.1E-06	NA
Plutonium-239	8.5E-06	1.8E-07
Tritium	(b)	(b)

a. NA indicates not available.
b. Although tritium emissions are not routinely monitored at these facilities, the releases from both basins were recently estimated as 1-2 Ci/year. These emissions could account for up to 25% of the total dose from these facilities to the maximally exposed offsite resident. However, the contribution from the 100 area tritium emissions would not change the estimated dose from all Hanford emissions to the site's maximally exposed offsite resident.

Table 5.7-2. Annual atmospheric releases for normal operation - fuel storage at 300 Area 308, 324, 325, and 327 buildings.

Radionuclide	308 Building Release (Ci/yr)	324 Building Release (Ci/yr)	325 Building Release (Ci/yr)	327 Building Release (Ci/yr)
Tritium	NA[a]	9.6E+00	2.5E+01	NA
Total beta[b]	1.1E-07	6.4E-07	2.4E-06	9.3E-07
Total alpha[c]	3.0E-08	3.9E-07	8.5E-07	1.1E-07

a. NA indicates not available.
b. Total beta emissions were assumed to be strontium-90 for modeling purposes.
c. Total alpha emissions were assumed to be plutonium-239 for modeling purposes.

Table 5.7-3. Annual atmospheric releases for normal operation - fuel storage at 200 West Area T Plant and 400 Area FFTF.

Radionuclide	200-West Area T Plant Release (Ci/yr)	400 Area FFTF Release (Ci/yr)
Argon-41	NA[a]	8.5E+00[b]
Total beta/strontium-90	1.2E-05	6.7E-06[c]
Cesium-137	1.3E-05	NA
Americium-241	2.0E-06	NA
Total alpha/plutonium-239	2.2E-05	1.1E-06[d]

a. NA indicates not available.
b. Releases of Ar-41 occurred during reactor operation in 1992. The reactor was subsequently shut down, and releases of short-lived activation products are not anticipated from future fuel storage activities.
c. Total beta emissions were assumed to be strontium-90 for modeling purposes.
d. Total alpha emissions were assumed to be plutonium-239 for modeling purposes.

offsite resident from all facilities combined would therefore be less than the sum of the individual facility receptor doses reported in Table 5.7-4. The peak collective dose to the population within 80 kilometers (50 miles) is 3E-2 person-rem per year, which is predicted to result in less than one fatal cancer (6×10^{-4}) over 40 years of storage.

5.7.1.2 Nonradiological Consequences. The No Action Alternative involves no new construction so there would not be an increase in particulate emissions. The facilities currently used in storing the SNF do not have any nonradiological releases, so there would be no increase in concentrations of these pollutants.

5.7.2 Decentralization Alternative

The Decentralization Alternative permits construction of new facilities where these represent an improvement over current storage practices. Relocation of fuel could be undertaken as part of this alternative to meet programmatic needs; however, no fuel would be shipped to, or received from, offsite locations. It is assumed for purposes of this analysis that new facilities would be constructed under this alternative, and that they would be located in a dedicated SNF management complex adjacent to the 200-East Area.

Table 5.7-4. Radiological consequences of airborne emissions during normal operation in the No-Action Alternative for spent nuclear fuel storage at Hanford.

		Onsite worker		Offsite resident		80 kilometer population	
Area	Facility	Peak annual dose (EDE) (rem/yr)	Probability of fatal cancer	Peak annual dose (EDE) (rem/yr)	Probability of fatal cancer	Peak annual dose (EDE) (person-rem/yr)	Number of fatal cancers
100 KE	Wet Basin	9.3E-06		2.0E-07		5.7E-03	
100 KW	Wet Basin	1.2E-07		3.3E-09		9.1E-05	
300	308 Bldg	3.3E-09		2.1E-09		1.4E-05	
300	324 Bldg	1.4E-08		2.9E-07		3.0E-03	
300	325 Bldg	1.2E-07		1.9E-06		1.1E-02	
300	327 Bldg	1.7E-09		2.4E-09		2.6E-05	
200 W	Burial Ground	0.0E+00		0.0E+00		0.0E+00	
200 W	T Plant	1.3E-07		3.3E-08		2.4E-03	
400	Fast Flux Test Facility	1.9E-06		1.9E-07		4.1E-03	
Total from All Facilities		1.2E-05	4.6E-09	2.6E-06	1.3E-09	2.7E-02	1.3E-05

The Decentralization Alternative at Hanford includes two basic options, each with several suboptions depending on the types of storage and processing facilities included. The first major option includes a combination of wet storage of defense production fuel and dry storage of all other fuel in either a small vault facility (suboption W) or in casks (suboption X). The second major option provides for dry storage of all fuel, which would require processing of defense fuel prior to dry storage. If a shear/leach/calcine process is used (suboption P), the calcine product and all other fuel would be consolidated in a single large vault facility (suboption Y) or in casks (suboption Z). If a solvent extraction process is chosen for the defense fuel (suboption Q), the oxide products could be stored in either new or existing facilities that would have lower space and shielding requirements than for the calcine product. A high-level liquid waste stream would also be produced and transferred to underground storage tanks. All fuel other than the processed defense fuel would be stored in a small vault facility or in casks as in suboptions W and X.

5.7.2.1 Radiological. Estimated radiological air emissions for normal operations of new facilities in the Decentralization Alternative are listed in Tables 5.7-5 through 5.7-7. The dry storage facilities are assumed to have no radiological emissions under normal operating conditions because all fuel is contained in sealed decontaminated canisters and storage casks. Therefore, there is no mechanism for routine release of radionuclides from dry storage facilities over the time period covered in this document.

The consequences of air emissions from individual facilities in the Decentralization Alternative are summarized in Table 5.7-8 and include a maximum annual dose of 2E-9 rem to a

Table 5.7-5. Estimated annual atmospheric releases for normal operation - new wet storage at 200-East Area.

Radionuclide	Release (Ci/yr)
Cobalt-60	1.4E-05
Strontium-90	1.1E-06
Ruthenium-106	6.2E-06
Cesium-137	2.3E-05
Europium-154	4.9E-06
Plutonium-238	1.1E-08
Plutonium-239	6.7E-08

Table 5.7-6. Estimated annual atmospheric releases for normal operation - shear/leach/calcine fuel process at 200-East Area.

Radionuclide	Release (Ci/yr)
Tritium	7.0E+02
Carbon-14	6.5E+00
Krypton-85	2.7E+05
Strontium-90	4.8E-07
Ruthenium-106	4.3E-09
Antimony-125	1.0E-08
Tellurium-125M	2.5E-09
Iodine-129	5.0E-03
Cesium-134	1.0E-08
Cesium-137	6.0E-07
Cerium-144	2.3E-09
Promethium-147	1.6E-07
Samarium-151	7.4E-09
Europium-154	7.2E-09
Americium-242	2.4E-12
Curium-242	6.1E-12
Plutonium-238	3.2E-09
Plutonium-241	3.8E-07
Americium-241	7.8E-09
Plutonium-239/240	1.5E-08

potential onsite worker (8E-13) probability of fatal cancer) for the option including a combination of wet and dry spent fuel storage facilities. The dose to an offsite resident at the highest exposure location is estimated as 6E-10 rem/year, and the corresponding probability of fatal cancer is 3E-13. The peak collective dose to the population within 80 kilometers is 2E-5 person-rem per year, which is predicted to result in less than one (4 x 10^{-7}) fatal cancer over 40 years of storage.

Table 5.7-7. Estimated annual atmospheric releases for normal operation - spent nuclear fuel solvent extraction fuel process at 200-East Area.

Radionuclide	Release (Ci/yr)
Tritium	7.0E+02
Carbon-14	6.5E+00
Krypton-85	2.7E+05
Strontium-90	2.4E-02
Ruthenium-106	5.1E-04
Antimony-125	4.6E-04
Tellurium-125M	2.4E-04
Iodine-129	1.9E-02
Cesium-134	5.1E-04
Cesium-137	3.0E-02
Cesium-144	1.2E-04
Promethium-147	8.1E-03
Samarium-151	7.4E-09
Europium-154	4.2E-04
Europium-155	1.7E-04
Americium-242	2.4E-12
Curium-242	6.1E-12
Plutonium-238	1.6E-03
Plutonium-241	1.9E-02
Americium-241	4.4E-03
Plutonium-239/240	8.0E-03

Table 5.7-8. Radiological consequences of airborne emissions during normal operation in the Decentralization Alternative for spent nuclear fuel storage at Hanford.

		Onsite worker		Offsite resident		80 km population	
Area	Facility	Peak annual dose (EDE) (rem/yr)	Probability of fatal cancer	Peak annual dose (EDE) (rem/yr)	Probability of fatal cancer	Peak annual dose (EDE) (person-rem/yr)	Number of fatal cancers
Combination Wet + Dry Storage Option							
200 E	New Wet Storage	2.0E-09	8.0E-13	5.7E-10	2.8E-13	2.3E-05	1.2E-08
200 E	New Dry Storage	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Dry Storage Only Option with Defense Fuel Processing							
200 E	New Dry Storage	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
200 E	New Fuel Calcine	4.1E-06	1.7E-09	7.0E-06	3.5E-09	3.4E-01	1.7E-04
200E	New Solvent Extraction	2.7E-05	1.1E-08	2.1E-05	1.1E-08	1.3E+00	6.3E-04

For the all dry storage option, processing defense fuel is required in the Decentralization Alternative (suboptions P and Q), and additional emissions would result from these activities if they were conducted. The dose to the onsite worker from air emissions would be 4E-6 rem per year for a shear/leach/calcine process or 3E-5 rem per year for a solvent extraction process (2E-9 or 1E-8 probability of fatal cancer, respectively) in addition to those from the dry storage facility. The corresponding consequences for the offsite resident would be 7E-6 rem per year (4E-9 probability of fatal cancer) for the shear/leach/calcine facility and 2E-5 rem per year (1E-8 probability of fatal cancer) for the solvent extraction facility. The collective dose to the offsite population from the respective fuel processing facilities is estimated at 0.3 to 1 person-rem per year, resulting in less than one expected fatal cancer (<0.02) over 40 years of storage.

5.7.2.2 Nonradiological Consequences. Fugitive dust emissions from new construction activities, toxic chemical emissions, and nitrogen oxide emissions from fuel processing would contribute to the nonradiological emissions in the Decentralization Alternative.

5.7.2.2.1 Fugitive Dust. Three different construction options are under consideration in this alternative: 1) construction of wet and dry storage facilities, 2) construction of dry storage and the shear/leach/calcine facility, and 3) construction of a dry storage and a solvent extraction facility. In options 1 and 2, approximately 12 acres would be disturbed for the construction of the storage facilities; in option 3, 6 acres would be disturbed for the dry storage facility. An additional 6 acres would be disturbed for the shear/leach/calcine facility or 12 acres for the solvent extraction facility. In total up to 12 acres would be disturbed in the first option and 18 acres in the second and third options (Bergsman 1995).

Details of the construction process are not available for the alternatives, but a standard default value of 1.2 tons/acre/month of particles can be assumed to be generated during new construction (EPA 1977). Most of the particles produced by construction activities are large and settle a short distance from the source (Seinfeld 1986). A conservative estimate is that approximately 30 percent of the mass released would be particles small enough to be transported away from the construction site (EPA 1988).

Experience with construction activities at Hanford indicates that fugitive dust concentrations at the nearest point of public access and at the site boundaries would be less than Washington State PM_{10} limits for both annual and 24-hour averages. Standard control techniques (such as applying water to the disturbed ground) could be used to limit the PM_{10}

emissions at the construction site and resulting airborne concentrations. Although extensive construction activities have the potential to contribute to short-term airborne particulate concentrations if they coincide with high wind events, such effects would generally be obvious only in the immediate area and could be mitigated by dust control measures over both the short and long term. In any case, such activities would be temporary and would not adversely affect regional air quality on a continuing basis. Construction activities would also result in increased emissions of pollutants from diesel- and gasoline-powered construction equipment. However, the increase in ambient levels of pollutants would be minimal because of the relatively low levels of emission and large distances to the nearest points of public access and the site boundary.

5.7.2.2.2 Nitrogen Oxides. Nitrogen oxide emissions during facility operation are approximately the same for both the shear/leach/calcine facility and the solvent extraction facility. It is assumed that all nitrogen oxide emissions are in the form of nitrogen dioxide. Annual concentrations at the nearest point of public access, 7.5 kilometers (6.4 miles) southwest of the release site, are estimated to be 0.1 micrograms per cubic meter. This concentration is 0.1 percent of the allowed Washington State standard and 0.4 percent of the Prevention of Significant Deterioration (PSD) standard.

Nitrogen oxide concentrations were also calculated for onsite locations. The maximum annual concentration estimated by the model is 1.2 micrograms per cubic meter, which occurs 500 meters (0.3 miles) south of the processing facility. The maximum ground level concentration is some distance from the processing facility because the emissions are from an elevated stack rather than at ground level. For example, at a distance of 100 meters (0.06 miles) from the base of the facility, the greatest estimated nitrogen oxide annual concentration is only 1.8×10^{-5} micrograms per cubic meter.

5.7.2.2.3 Toxic Chemical Emissions. Information about routine toxic chemical emissions from either the shear/leach/calcine facility or the solvent extraction facility is unavailable. However control techniques would be used to ensure that concentrations of toxics in the atmosphere comply with the DOE abatement policy and local permitting requirements.

5.7.3 1992/1993 Planning Basis Alternative

The 1992/1993 Planning Basis Alternative is assumed to be similar to the Decentralization Alternative discussed in the previous section, including construction of wet or dry storage facilities adjacent to the 200-East Area and process facilities for defense production fuel if it is to be stored dry. The only change to the Hanford Site fuel inventory would involve shipment of a relatively small quantity of TRIGA fuel to an offsite location. This would not substantially alter the scope of planned spent fuel storage activities, and the 1992/1993 Planning Basis Alternative assumes emissions for new facilities are the same as those in the Decentralization Alternative.

5.7.3.1 Radiological Consequences. The consequences for this alternative are assumed to be the same as those for the Decentralization Alternative. Refer to Table 5.7-8 for the list of facilities included in this option and their consequences.

5.7.3.2 Nonradiological Consequences. The consequences for this alternative are considered to be the same as those for the Decentralization Alternative.

5.7.4 Regionalization Alternative

The Regionalization Alternative at Hanford includes three options, depending on the quantity of SNF shipped to, or from, the site. Option A provides for regional storage of SNF by type, and would entail shipping all fuel at Hanford except defense production fuel to another location. In this case, defense fuel would either be stored wet at a new pool facility, or it would be processed for dry storage using suboptions similar to those described in the Decentralization Alternative.

An additional option in the Regionalization Alternative describes importing SNF to Hanford from other sites based on their geographic distribution. In the first option, designated Option B1, all fuel at locations west of the Mississippi River except Naval SNF would be stored at Hanford. In the second option, designated Option B2, all SNF at locations west of the Mississippi River and Naval SNF would be stored at Hanford. All imported fuel would ultimately be placed into a new dry storage facility, the size of which would be determined by the quantity of imported fuel to be stored. In addition, a receiving and canning facility would be built to repackage any fuel as needed, and to provide temporary wet storage for fuels that could not be immediately placed into dry storage. This option would also include a technology development facility for fuel characterization and research related to SNF management. SNF currently at Hanford would be stored according to the options described in the Decentralization

Alternative. Option B2 would include a separate facility to examine and characterize Naval SNF, as described in Appendix D to Volume 1 of this EIS.

The third Regionalization option (designated Option C) would relocate all SNF at the Hanford Site to another western U.S. location. The only new facility that would be required for this option is a processing and packaging facility to stabilize and repackage defense fuel and to place other fuel into canisters as needed for shipping offsite. Prior to preparation for offsite shipment, SNF would continue to be managed at existing facilities, as for the No Action Alternative. All new facilities considered in the Regionalization Alternative options would be constructed in a dedicated SNF management complex adjacent to the 200-East Area, as for the Decentralization Alternative.

5.7.4.1 Radiological Consequences. Emissions from new facilities in Regionalization Alternative A would be the same as those described for the Decentralization Alternative in Table 5.7-8. Although this option does not include the dry storage capacity for fuel other than defense production fuel, dry storage facilities add nothing to the normal operating emissions; therefore, the emissions and consequences from this alternative would be quantitatively the same as those previously described for the Decentralization Alternative.

Emissions from the new facilities in the Regionalization Alternative B and C options are expected to be bounded by those in the Centralization maximum and minimum options, respectively, as described in Section 5.7.5.

5.7.4.2 Nonradiological Consequences. Because of the similarity of operations, consequences for the Regionalization Alternative are considered to be the same as those for the Decentralization Alternative.

5.7.5 Centralization Alternative

The Centralization Alternative at Hanford includes two options: a maximum option in which all SNF for which DOE is responsible would be stored at Hanford, and a minimum option in which all SNF currently at Hanford would be shipped to another site. The maximum option is similar to that described in the Regionalization Option B2, except that the size of the receiving and canning and dry storage facilities would be increased as necessary to accommodate the larger quantity of imported fuel. The minimum option is identical to that described for the

Regionalization Alternative, Option C. All new facilities considered in the Centralization Alternative options would be constructed in a dedicated SNF management complex adjacent to the 200-East Area.

5.7.5.1 Radiological. For the Centralization maximum option at Hanford, emissions from the wet storage and processing facilities would be identical to those described in the Decentralization Alternative (refer to Tables 5.7-5 through 5.7-7). Minimal emissions from the large dry storage facility are assumed in this case (see Table 5.7-9) because some of the imported fuel could be stored without canning, and the assumption of zero emissions could not be justified as in the Decentralization Alternative. The consequences of emissions from a relocated Expended Core Facility (ECF) are described in Appendix D to Volume 1 of this EIS and are not included here. It should be noted that the assumptions used in Appendix D calculations for the ECF at Hanford may differ from those used to estimate the consequences of emissions from other Hanford facilities.

The consequences of air emissions from individual facilities in the Centralization Alternative maximum option are summarized in Table 5.7-10 and include a maximum annual dose of 9E-9 rem to a potential worker (4E-12 probability of fatal cancer) for a combination of wet and dry spent fuel storage facilities. The dose to an offsite resident at the highest exposure location is estimated as 2E-9 rem/year, and the corresponding probability of fatal cancer is 8E-13. The peak collective dose to the population within 80 kilometers is 7E-5 person-rem per year, which is predicted to result in less than one (4×10^{-8}) fatal cancer.

Table 5.7-9. Estimated annual atmospheric releases for normal operation - new dry storage at 200-East Area (maximum option).

Radionuclide	200-East Area Release (Ci/yr)
Cobalt-60	2.8E-08
Strontium-90	9.1E-07
Yttrium-90	9.1E-07
Cesium-137	1.2E-07
Plutonium-239	2.8E-07

Table 5.7-10. Radiological consequences of airborne emissions during normal operation in the Centralization Alternative for spent nuclear fuel storage at Hanford.

		Onsite worker		Offsite resident		80 km population	
Area	Facility	Peak annual dose (EDE) (rem/yr)	Probability of fatal cancer	Peak annual dose (EDE) (rem/yr)	Probability of fatal cancer	Peak Annual Dose (EDE) (Person-rem/yr)	Number of Fatal Cancers
Combination Wet + Dry Storage Option							
200 E	New Wet Storage	2.0E-09	8.0E-13	5.7E-10	2.9E-13	2.3E-05	1.2E-08
200 E	New Dry Storage	7.0E-09	3.0E-12	1.0E-09	5.0E-13	4.8E-05	2.4E-08
Dry Storage Only Option with Defense Fuel Processing							
200 E	New Dry Storage	7.0E-09	3.0E-12	1.0E-09	5.0E-13	4.8E-05	2.4E-08
200 E	New Fuel Calcine	4.1E-06	1.7E-09	7.0E-06	3.5E-09	3.4E-01	1.7E-04
200E	New Solvent Extraction	2.7E-05	1.1E-08	2.1E-05	1.1E-08	1.3E+00	6.3E-04
Relocation of Expended Core Facility[a]							

a. Data for the expended core facility (ECF) are presented in Appendix D to Volume 1 of this EIS. Assumptions used in Appendix D calculations for the ECF at Hanford may differ from those used to estimate the doses consequences of emission from other Hanford facilities.

Processing of defense fuel is required prior to dry storage in the maximum option, and additional air emissions would result from those activities if defense fuel is stored dry rather than wet. The dose to the worker would increase by 4E-6 rem/year for a shear/leach/calcine process or 3E-5 rem/year for a solvent extraction process (2E-9 or 1E-8 probability of fatal cancer, respectively). The corresponding added consequences for the offsite resident would be 7E-6 rem/year (4E-9 probability of fatal cancer) for the shear/leach/calcine facility and 2E-5 rem/year (1E-8 probability of fatal cancer) for the solvent extraction facility. The collective dose to the offsite population from the respective fuel processing facilities is estimated at 0.3 to 1 person-rem per year, resulting in less than one (5×10^{-4}) fatal cancer.

In the Centralization Alternative minimum option, the consequences of existing facilities utilized for interim fuel storage prior to shipment offsite are the same as in the No Action Alternative. Consequences for defense fuel processing prior to shipment are described under the centralization maximum alternative and are equivalent to those from the shear/leach/calcine facility. Refer to Tables 5.7-4 and 5.7-10 for the consequences of facilities included in this option.

5.7.5.2 Nonradiological. Because of the similarity of operations leading to nonradiological impacts on air quality, consequences for the Centralization Alternative are considered to be the same as those for the Decentralization Alternative with the addition of emissions from the naval fuels Expended Core Facility. Analysis of nonradiological releases from the Expended Core Facility can be found in Volume 1, Appendix D.

5.8 Water Quality and Related Consequences

This section evaluates the potential impacts to groundwater and surface water resources from the construction and operation of SNF storage and associated support facilities at the Hanford Site. Potential impacts to groundwater and surface water, water use, and water quality from the potential release of contaminants into, and migration through, hydrologic water-based environments are evaluated. The potential significance of these impacts is evaluated with respect to environmental contaminant levels from potential releases of contaminants into the environment and the health impacts of these contaminant levels. Contaminant waste streams include radionuclide and chemical carcinogens and noncarcinogenic chemicals.

The Multimedia Environmental Pollutant Assessment System (MEPAS), a computer model, was utilized to simulate the release, migration, fate, exposure, and risk to surrounding receptors of wastes that are discharged into the environment from the operation of SNF facilities. The MEPAS model is a fully integrated, physics-based, PC-platform, intermedia transport- and risk-computation code that is used to assess health impacts from actual and potential releases of both hazardous chemicals and radioactive materials. The MEPAS model is designed for site-specific assessments using readily available information. It follows EPA risk-assessment guidance in evaluating 1) the release of contaminants into the environment; 2) their movement through and transfer between various environmental media [i.e., subsurface (vadose and saturated zones), surface water, overland (surface soil), and atmospheric]; 3) exposure to surrounding receptors via inhalation, ingestion, dermal contact, and external dose; and 4) risk to carcinogens and hazard to noncarcinogens. The MEPAS model follows ICRP/NCRP and EPA guidelines, where the user is allowed to choose the appropriate guidelines.

5.8.1 No Action Alternative

The only release directly to the surface water in the No Action Alternative was associated with the 105-KE and 105-KW basins. The 105-KE and 105-KW basins were combined as one release and represented by a "single liquid release point to the Columbia River" (Bergsman 1995). The annual liquid discharge is assumed to be 1.4E+06 cubic meters per year (3.7E+08 gallons per year), with a total activity of approximately 0.4 Ci: 0.26 Ci tritium, 0.066 Ci cobalt-60, 0.01 Ci cesium-137, 0.0010 Ci strontium-90, and 9.2E-06 Ci plutonium-239 (Bergsman 1995). All of the constituents in this assessment are radionuclides. The release is assumed to continue at this level over the period of 18 years from 1997 through 2015. Operational liquid effluents from the K Basins are discharged to the Columbia River via the monitored and regulated National Pollutant Discharge Elimination System (NPDES) permitted 1908-KE outfall. Contaminant migration is from the point-source discharge point to the Columbia River, and in the Columbia River to receptors downstream. The flow discharge in the Columbia River is assumed to be under low-flow conditions of 1,000 cubic meters per second (36,000 cubic feet per second) (Whelan et al. 1987), which represents the most conservative case for maximizing surface water concentrations. As a conservative assumption, the removal of water from the Columbia River is assumed to be 100 meters (328 feet) downstream of the point of entry of the contaminant into the river. The assessment addressed recreational activities (e.g., boating, swimming, and fishing) in the Columbia River and use of the water as a drinking water supply and for bathing, irrigation, etc. The risk of fatal cancer in this scenario considering all pathways was found to be less than one chance in a billion. For more information, refer to Whelan et al. (1994).

Intermittent leakage of water from the K Basins is monitored via onsite groundwater sampling. Although radionuclide concentrations in some of the 100-K area monitoring wells exceed EPA drinking water standards, this condition does not constitute a risk to the public because the groundwater is not used directly for human consumption or food production. Analyses of water from the K area springs, where groundwater enters the Columbia River, indicate that radionuclide levels are below the EPA drinking water standards. Dilution of this seepage in the river flow would further reduce the risk to the downstream population, as indicated by the fact that radionuclide concentrations in the Columbia River at the Richland pump house are orders of magnitude below the drinking water standard (Dirkes et al 1994).

5.8.2 Decentralization Alternative

The Spent Nuclear Fuel Wet Transfer and Storage scenario was documented. The source term represents the maximum potential water releases that would be expected if a secondary containment failure and/or piping leak occurred and went undetected for one month at a state-of-the-art wet storage fuel/transfer facility utilizing water treatment technology now available. Releases resulting from such a failure should not be thought of as operational or planned releases. However, for the purposes of a nonzero release source-term, this scenario addresses those situations where an unexpected release may occur. The source-term information was derived from data related to the operation of the Flourinel and Storage Facility (FAST) at INEL's Chemical Processing Plant (ICPP 666) and is considered to be extremely conservative, given the state-of-the-art engineering practices, monitoring, leak-detection equipment, and surveillance procedures likely to be used at any new SNF facility, such as FAST.

Any new facility would be built using state-of-the-art technologies, including leak detection and water-balance monitoring equipment. This equipment, along with the uncertainties associated with evaporation monitoring, will have a minimum detection sensitivity. It is possible that the new SNF facility could experience a failure that would result in a leak that is below the sensitivity of the detection system. Based on the size of the facility and the current monitoring programs at similar facilities, 5 gallons per day has been established as a conservative value to account for potential undetected leakage from the facility. The nonzero release source term would then exceed what could be expected for a new SNF wet storage or transfer facility. Factors contributing to the conservatism in volume estimates are the design criteria, which state that the new facility will contain leak-detection systems (Hale 1994) and will have a lower surface area [i.e., 2000 square meters (6600 square feet)] available for leakage as compared to FAST [i.e., 3830 square meters (12,560 square feet)] (Hale 1994). For the purposes of this assessment, the entire release is assumed as a point source, which is the most conservative assumption. The concentration data associated with the release were contained in or derived from January 6, 1986 to February 14, 1994 weekly water quality reports for FAST and are considered to be reasonable nonzero release source terms at the 95% confidence level. Although surveillance at the FAST facility occurs daily with radiological surveys occurring weekly, the aqueous release assumes that the liner and/or piping leaks and secondary containment failure go undetected for one month.

The specific radionuclide activities in the release solution are assumed as follows: 280 pCi/L strontium-90, 3360 pCi/L cobalt-60, 160 pCi/L cobalt-57[b], 93 pCi/L cesium-137, and 100 pCi/L antimony-125. All of the constituents in this assessment are radionuclides. Contaminant migration is through the vadose zone through the saturated zone to the Columbia River, and in the Columbia River to receptors downstream. The flow discharge in the Columbia River is assumed to be under low-flow conditions 1000 m^3 per second (36,000 cubic feet per second) (Whelan et al. 1987), which represents the most conservative case for maximizing surface water concentrations. As a conservative assumption, the removal of water from the Columbia River is assumed to be 100 meters (328 feet) downstream of the contaminant influent point to the river. The assessment addresses recreational activities (e.g., boating, swimming and fishing) in the Columbia River and use of the water as a drinking-water supply and for bathing, irrigation etc. The risk of fatal cancers considering all pathways was found to be significantly less than one chance in a trillion. For more information, refer to Whelan et al. (1994).

The Decentralization Alternative also includes an operational release scenario to the Hanford 200 Area Treated Effluent Disposal Facility (TEDF). Liquid effluents would be added to the TEDF, which receives liquid effluent from many facilities in the 200 Area. The "Discharge Target" allowable concentrations in the TEDF are presented in Bergsman (1995). Only 380 liters (100 gallons) per day will be discharged to the TEDF basin from this operation,

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a. Cobalt-57 is substituted in the analysis for cobalt-58 because the MEPAS database contains only cobalt-57.

although other facilities unrelated to SNF storage will also be discharging to the basin. For a ponded situation, the maximum outflow from the basin is equal to the transmission rate (i.e., saturated hydraulic conductivity under a unit hydraulic gradient) of the soil immediately below the basin, which is 24 cubic meters per day (6260 gallons per day). To maximize the flow velocity through the vadose zone and the mass flux of contaminant leaving the basin (i.e., concentration x area x flow velocity), the assessment assumes that this facility leaks into the vadose zone over a 4-year period with the infiltration rate limited by the transmission rate of the soil. The discharge from the pond is assumed to last for 4 years from 2002 through 2006.

Based on the movement of the second tritium plume from the Plutonium and Uranium Recovery through Extraction cribs in the 200 Area to Well 699-24-33, a distance of 6 kilometers (4 miles) in a 5-year period (1983 to 1988), the average pore-water velocity (i.e., specific discharge divided by the effective porosity) in the saturated zone was 3.3 meters per day (10.8 feet per day) (Schramke et al. 1994). Davis et al. (1993) performed a more recent analysis and determined the pore-water velocity as 0.02 meters per day (0.08 feet per day) just below the TEDF site, although this is not necessarily indicative of the velocity as the water moves toward the river. Both velocities were initially used in assessing the migration of contamination from the basin to determine the most conservative result with respect to risk. In the final analysis, the highest pore-water velocity of 3.3 meters per day (10.8 feet per day) was used because 1) it is consistent with other assessments at the installation, 2) the contaminants reached the river and receptors earlier, and 3) the resulting exposure analysis provided the more conservative estimate of risk over the 7000-year assessment time frame.

Radionuclides, chemical carcinogens, and noncarcinogens are contained in the waste stream. The concentrations in the TEDF were represented by the discharge target allowable concentrations. Contaminant migration is from the ponded water, through the vadose zone, through the saturated zone to the Columbia River, and in the Columbia River to receptors downstream. The flow discharge in the Columbia River is assumed to be under low-flow conditions of 1000 cubic meters per second (36,000 cubic feet per second) (Whelan et al. 1987), which represents the most conservative case for maximizing surface water concentrations. As a conservative assumption, the removal of water from the Columbia River is assumed to be

100 meters (328 feet) downstream of the point of entry of the contaminant into the river. The assessment addressed recreational activities (e.g., boating, swimming, and fishing) in the Columbia River and use of the water as a drinking-water supply and for bathing, irrigation, etc.

The maximum radionuclide and chemical carcinogenic risks were found to be less than 50 chances in a billion for all of the constituents through all of the exposure routes. Likewise, noncarcinogenic chemical individual doses were found to be below their respective reference doses, except chromium VI, which had a dose about 50 percent higher than the reference dose. Chromium VI had an assigned distribution coefficient (i.e., Kd) of zero (Serne and Wood 1990), which represents the most mobile condition in the vadose zone. For more information, refer to Whelan et al. (1994).

5.8.3 1992/1993 Planning Basis Alternative

Scenarios and consequences relating to water quality would be the same as for the Decentralization Alternative. For more information, refer to Whelan et al. (1994).

5.8.4 Regionalization Alternative

Scenarios and consequences relating to water quality in the Regionalization options would be the same as for water quality aspects in the Decentralization Alternative. For more information, refer to Whelan et al. (1994).

5.8.5 Centralization Alternative

Scenarios and consequences relating to water quality would be the same as for the Decentralization Alternative. For more information, refer to Whelan et al. (1994).

5.9 Ecological Resources

Implications of implementing the alternatives for interim storage of SNF on terrestrial resources, wetlands, aquatic ecosystems, and threatened and endangered species at the Hanford Site are discussed in the following subsections.

5.9.1 No Action Alternative

Implications of implementing the No Action Alternative for interim storage of SNF on terrestrial resources, wetlands, aquatic resources, and threatened and endangered species at the Hanford Site are discussed in the following subsections.

5.9.1.1 Terrestrial Resources. No new SNF facilities would be constructed at Hanford and there would be no impacts to the terrestrial resources of the Hanford Site beyond those resulting from natural processes of succession and the impacts of ongoing Hanford operations. They would remain as described under Section 4.9.1.

5.9.1.2 Wetlands. No new SNF facility would be constructed; therefore, no changes to wetlands on the Hanford Site would be expected beyond those changes resulting from natural processes and the impacts of ongoing Hanford operations (see Section 4.9.3).

5.9.1.3 Aquatic Resources. No new SNF facility would be constructed and the fact that there are no surface water facilities on the SNF facility site indicates that there would be no impacts on the aquatic resources of the Hanford Site other than those changes resulting from natural processes and the impacts of ongoing Hanford operations and they would remain as described in Section 4.9.3.

5.9.1.4 Threatened and Endangered Species. No new SNF facilities would be constructed and operated at Hanford. Thus, populations of species listed as endangered or threatened, or candidates for such listing by the federal and Washington State governments, or species listed as monitor species by the Washington State government would not be impacted (either directly by displacement or indirectly by habitat alteration) beyond effects resulting from ongoing Hanford operations and natural processes.

5.9.1.5 Radioecology. Releases of radionuclides to the environment are expected to be on the order of those released in the recent past by site operations (Woodruff and Hanf 1993), and thus will not be accumulated into terrestrial or aquatic ecosystems in concentrations that could cause measurable impacts.

5.9.2 Decentralization Alternative

Implications of implementing the Decentralization Alternative for interim storage of SNF on terrestrial resources, wetlands, aquatic resources, and threatened and endangered species at the Hanford Site are discussed in the following subsections.

5.9.2.1 Terrestrial Resources. This alternative would require the construction of an SNF facility for fuel management and storage. Most spent fuel from the Hanford Site would be stored here.

Construction of an SNF facility at Hanford would disturb up to 9 hectares (24 acres) on the 65 hectare (160 acres) site, representing about 0.01 percent of the total area of the Hanford Site. Approximately 9 hectares (24 acres) would be occupied by facilities, access roads, or rights-of-way and therefore, would remain developed for the life of the project. The remaining land would be revegetated with native grasses and shrubs upon completion of construction.

Vegetation within construction areas would be destroyed during land-clearing activities. Plant species that are dominant on the Hanford SNF site, and thus would be most affected, include big sagebrush, cheatgrass, and Sandberg's bluegrass. Total area destroyed would amount to about less than 1 percent of this community on the Hanford Site. Although the plant communities to be disturbed are well-represented on the Hanford Site, they are relatively uncommon regionally because of the widespread conversion of shrub-steppe habitats to agriculture. Disturbed areas are generally recolonized by cheatgrass, a nonnative species, at the expense of native plants. Mitigation of these impacts could include minimizing the area of disturbance and revegetating with native species, including shrubs, and establishing a 2:1 acreage replacement habitat in concert with a habitat enhancement plan presently being developed for the Hanford Site in general. Adverse impacts to vegetation on Hanford are expected to be limited to the project area and vicinity and are not expected to affect the viability of any plant populations on the Hanford Site.

Construction of an SNF facility and support facilities would have some adverse affect on animal populations. Less mobile animals such as invertebrates, reptiles, and small mammals within the project area would be destroyed during land-clearing activities. Larger mammals and birds in construction and adjacent areas would be disturbed by construction activities and would move to adjacent suitable habitat, and these individual animals might not survive and reproduce.

Project facilities would displace about 9 hectares (up to 24 acres) of animal habitat for the life of an SNF facility. Revegetated areas (e.g., construction laydown areas and buried pipeline routes) would be reinvaded by animal species from surrounding, undisturbed habitats. The adverse impacts of construction are expected to be limited to the project area and vicinity and should not affect the viability of any animal populations on the Hanford Site because similar suitable habitat would remain abundant on the site.

Very small quantities of radionuclides would be released to the atmosphere during SNF facility operations. No organisms studied to date are reported to be more sensitive than man to radiation (NRC-8). Therefore, as concluded for humans, the effects of these releases on terrestrial organisms are expected to be minor.

These impacts to the vegetation and animal communities could be mitigated by minimizing the amount of land disturbed during construction, employing soil erosion control measures during construction activities, and revegetating disturbed areas with native species. These measures would limit the amount of direct and indirect disturbance to the construction area and surrounding habitats and would speed the recovery process for disturbed lands.

Operational impacts to terrestrial biotic resources would include exposure of plants and animals to small amounts of radionuclides released during operation of the SNF facility. The levels of radionuclide exposure would be below those levels that produce adverse effects.

5.9.2.2 Wetlands. No wetlands occur on or near the SNF facility site, so no impacts from the construction and operation of the facility to wetlands would occur. Wetlands resources on the Hanford Site would remain as described in Section 4.9.2. No mitigation efforts would be required because no wetlands would be affected.

5.9.2.3 Aquatic Resources. No aquatic habitats occur on the SNF site; thus, no impacts to aquatic resources are expected from the construction and operation of the SNF facility. No mitigation efforts would be required because no impacts are anticipated to aquatic resources.

5.9.2.4 Threatened and Endangered Species. Construction and operation of the SNF facility would remove approximately 9 hectares (24 acres) of relatively pristine big sagebrush/cheatgrass-Sandberg's bluegrass habitat. This sagebrush habitat is considered priority habitat by the State of Washington because of its relative scarcity in the state and its use as nesting/breeding habitat by loggerhead shrikes, sage sparrows, sage thrashers, burrowing owls, pygmy rabbits, and sagebrush voles. Bald Eagles, peregrine falcons, and Oregon silverspot butterflies do not inhabit the potential proposed site.

Loggerhead shrikes, listed as a federal candidate (Category 2) and state candidate species, forage on the proposed SNF site and are relatively common on Hanford. This species is sagebrush-dependent, as it is known to select primarily tall big sagebrush as nest sites. Construction of the SNF facility would remove big sagebrush habitat which would preclude loggerhead shrikes from nesting there. SNF site development would also be expected to reduce the value of the site as foraging habitat for shrikes known to nest in adjacent areas.

Sage sparrows and sage thrashers, both state candidate species, occur in mature sagebrush/bunchgrass habitat at Hanford. Sage thrashers were not observed on the SNF site, and are extremely rare on the Hanford Site. These species are known to nest primarily in sagebrush. Construction of the SNF facility would preclude both of these species nesting there and reduce the site's suitability as foraging habitat for these species.

SNF construction is not expected to substantially decrease the Hanford population of loggerhead shrike, sage sparrow, or sage thrashers because similar sagebrush habitat is still relatively common on the Hanford Site. However, the cumulative effects of constructing the SNF facility, in addition to future developments that further reduce sagebrush habitat (causing further fragmentation of nesting habitat), could negatively affect the long-term viability of populations of these species on the Hanford Site.

Burrowing owls, a state candidate species, are relatively common on the Hanford Site and nest in abandoned ground squirrel burrows on the proposed SNF site. SNF construction would remove sagebrush and disturb soil, displacing ground squirrels and thus reducing the suitability of the area for nesting by burrowing owls. Construction would also displace small mammals, which constitute a portion of the prey base for this species. Construction for an SNF facility would, however, not be expected to negatively impact the viability of the population of burrowing owls on Hanford, as their use of ground squirrel burrows as nests is not limited to burrows in big sagebrush habitat.

Pygmy rabbits, a federal candidate (Category 2) and state threatened species, are known to utilize tall clumps of big sagebrush habitat throughout most of their range. However, this species has not recently been observed on the Hanford Site. Construction of the SNF facility would therefore reduce the potential for recolonization by this species by removing habitat suitable for its use.

Sagebrush voles, a state monitor species, are common on the Hanford Site and select burrow sites near sagebrush; however, this species is common only at higher elevations around the Hanford Site. Construction of the SNF facility would remove sagebrush habitat, precluding sagebrush voles from utilizing the site. However, construction would not affect the overall viability of sagebrush vole populations on the Hanford Site because the majority of the population is found on the Fitzner/Eberhardt Arid Lands Ecology Preserve.

The closest known nests of ferruginous hawks, a federal candidate (Category 2) and state threatened species, and Swainson's hawk, a state candidate, are 8.5 km (5 mi) and 6.2 km (3.7 mi), respectively, from the proposed SNF site. The SNF site comprises a portion of the foraging range of these hawks. Construction of the SNF facility is not expected to disrupt the nesting activities of these species. However, construction would displace small mammal populations and thus reduce the prey for these birds. The cumulative effects of constructing the SNF facility, in addition to future reductions in sagebrush habitat (causing further fragmentation of foraging habitat), could negatively affect the long-term viability of populations of these two species on Hanford.

5.9.2.5 Radioecology. Releases of radionuclides to the environment are expected to be below those currently released by site operations (Woodruff and Hanf 1993), and thus will not be accumulated into terrestrial or aquatic ecosystems in concentrations that could cause measurable impacts.

5.9.3 1992/1993 Planning Basis Alternative

The 1992/1993 Planning Basis Alternative differs from the Decentralization Alternative only in that TRIGA fuel currently stored at the Hanford Site would be shipped to INEL for storage. (It is possible that the TRIGA fuel may be transferred to third parties for beneficial use prior to the planned time of shipment to INEL.) Thus, impacts on terrestrial resources, wetlands, aquatic resources, threatened and endangered species, and radioecology at the Hanford Site would be essentially the same as described for the Decentralization Alternative.

5.9.4 Regionalization Alternative

All new facilities would be constructed on the 65 hectare (163-acre) site west of 200-East Area (Figure 4.1). Although impacts on terrestrial resources are expected to be minimal, the impacts that would occur would be roughly proportional to the amount of land that would be disturbed during construction. For the various options of the Regionalization Alternative, those areas would amount to the following amounts of land:

A) From about 2 to 7 hectares (5 to 18 acres) when all SNF except defense production SNF would be sent to INEL.

B1) From about 15 to 17 hectares (38 to 43 acres) when all SNF west of the Mississippi River except Naval SNF would be sent to Hanford.

B2) From about 25 to 28 hectares (63 to 70 acres) when all SNF west of the Mississippi River and Naval SNF would be sent to Hanford.

C) From about 2 to 5 hectares (5 to 12 acres) when all Hanford SNF would be sent to INEL or NTS.

While the largest area cited above (28 hectares) is about three times the size of the area to be disturbed in the Decentralization Alternative, it is still a very small fraction of similar habitat on the Hanford Site. By and large the discussion on flora and fauna presented in Section 5.9.2 applies to the Regionalization Alternative, bearing in mind that the area involved would be more or less depending on the option chosen.

5.9.5 Centralization Alternative

If Hanford is selected as the site for the Centralization Alternative, an SNF facility, as substantially described in the Decentralization Alternative, would be constructed at Hanford.

Although the facility would store about 25 weight percent more SNF than would be stored under the Decentralization Alternative and the number of casks would increase the required space, the ecological impacts would be essentially the same as those described in Section 5.9.2.

If Hanford is not selected as the site for the Centralization Alternative, an SNF packaging facility would be built to prepare the fuel for shipment offsite. While that facility would not be as extensive as the SNF facility, the ecological impacts would not likely be importantly different from those described in Section 5.9.3 for the Decentralization Alternative.

5.10 Noise

Implications of implementing the alternatives for interim storage of SNF on noise levels at the Hanford are discussed in the following subsections.

5.10.1 No Action Alternative

Under this alternative, new SNF facilities would not be constructed, and the noise associated with SNF facility construction and operation activities would not occur. Because no major changes in existing noise-emitting sources are expected at Hanford during the projected SNF facility construction period, the ambient noise levels at Hanford would be expected to remain essentially the same for the no-action alternative as during the baseline period.

5.10.2 Decentralization Alternative

This alternative would require the construction and operation of an SNF facility for fuel management and storage. Most spent fuel from the Hanford Site would be stored here. The results of a detailed analysis of the potential noise impacts from constructing and operating a new production reactor (project since cancelled) and its support facilities at Hanford have been published. The analysis indicates that noise from constructing a facility the size of a production reactor, and from operational facilities, equipment, and machines, would not cause ambient noise levels to exceed the limits set by the Washington State noise control regulations or EPA guidelines. The latter are set to protect the public from the effect of broadband environmental noise and to protect the public against hearing loss. The results also indicate that increases in noise levels from constructing and operating a facility the size of a production reactor and its support facilities, including increased traffic along the major roadways, would result in little or no increase in the annoyance level experienced by communities or individuals.

No significant noise impacts from activities associated with SNF facility construction and operation are expected at sensitive receptor locations outside the Hanford boundary or at residences along the major highways leading to the proposed SNF site at Hanford.

5.10.3 1992/1993 Planning Basis Alternative

The 1992/1993 Planning Basis Alternative differs from the Decentralization Alternative only in that TRIGA fuel currently stored at the Hanford Site would be shipped to INEL for storage. (It is possible that the TRIGA fuel may be transferred to third parties for beneficial use prior to the planned time of shipment to INEL.) Thus, impacts would be essentially the same as described for the Decentralization Alternative.

5.10.4 Regionalization Alternative

All new facilities would be constructed on the 65 hectare (163-acre) site west of 200-East Area (Figure 4.1). Although noise is not expected to be a factor in evaluating the alternatives, the amount and duration of noise associated with construction would be roughly proportional to the amount of land that would be disturbed during construction. For the various options of the Regionalization Alternative, those areas would amount to the following amounts of land:

A) From about 2 to 7 hectares (5 to 18 acres) when all SNF except defense production SNF would be sent to INEL.

B1) From about 15 to 17 hectares (38 to 43 acres) when all SNF west of the Mississippi River except Naval SNF would be sent to Hanford.

B2) From about 25 to 28 hectares (63 to 70 acres) when all SNF west of the Mississippi River and Naval SNF would be sent to Hanford.

C) From About 2 to 5 hectares (5 to 13 acres) when all Hanford SNF would be sent to INEL or NTS.

Although not likely to be heard offsite, the duration of noise that is generated would range from about a quarter to three times that described for the Decentralization Alternative depending on the Regionalization option chosen.

5.10.5 Centralization Alternative

If Hanford is selected as the site for centralization of SNF, new SNF facilities would be constructed at Hanford. Although somewhat larger than for the Decentralization Alternative, the impacts from noise would be the same as those described in Subsection 5.10.2.

5.11 Traffic and Transportation

The implications of implementing the alternatives for interim storage of SNF on traffic and incident-free onsite transportation of SNF and materials supporting SNF storage at the Hanford are discussed in the following subsections. The impacts of offsite transportation of SNF are discussed in Appendix I.

5.11.1 No Action Alternative

Implications of implementing the No Action Alternative for interim storage of SNF on traffic and incident-free onsite transportation of SNF and materials supporting SNF storage are discussed in the following subsections.

5.11.1.1 Traffic. Under the No Action Alternative, the number of workers would stay the same as under present conditions; therefore, there would be no change in traffic patterns. At present, there are periods of moderate traffic congestion, some of which is expected to be alleviated by a new road to the 200 areas.

5.11.1.2 Transportation. The RISKIND (Yuan et al. 1993) and RADTRAN 4 (Neuhauser and Kanipe 1992) computer codes were applied to calculate the radiation doses to transport workers and the public that are estimated to result from incident-free onsite transportation of SNF. RISKIND was also used to calculate the consequences of bounding transportation accidents. All of the onsite SNF shipments were assumed to emit radiation that would result in a dose rate at the regulatory limit (i.e., 0.01 rem per hour at 2 meters (6 feet) from the external surface of the shipments). This assumption contributes to the conservatism of the analysis because the shipment dose rates cannot be larger than this value but frequently will be substantially smaller. All shipments were assumed to be made by truck. A detailed description of the approach and other important shipment-related parameters are discussed in Volume 2, Chapter 5, and Appendix I. Hanford-specific information and input parameters are presented in this section.

The doses per incident-free shipment of each type of SNF were calculated using RISKIND and RADTRAN 4. The potential receptors considered are the transportation crew of two, on-link (on the road) and off-link (persons near the roadway) populations. Guards and/or inspectors may also be exposed to the shipments. Guards and inspectors may be exposed when they prepare a shipment to leave its origin facility or prepare to receive a shipment that has arrived at a destination facility. Guards and inspectors may also be exposed while the shipment is enroute between facilities. Guard and inspector doses at origin and destination facilities are included in the doses calculated in Section 5.13. Most onsite shipments originate in the 200 and 100 Areas and will not travel through a guarded checkpoint. The guard/inspector doses for these shipments are zero. Only the miscellaneous fuel shipments originating in the 300 Area and the FFTF shipments originating in the 400 Area will travel past a guarded checkpoint (see Wye Barricade in Section 4.11). Doses to the guards at the Wye Barricade were calculated assuming they were exposed briefly at a distance of 5 meters, (16 feet) from the shipment, as described in Volume 2, Chapter 5. The computer code RISKIND was used to calculate maximum and individual doses; RADTRAN 4 was used to calculate collective population doses.

Five general classes of SNF were considered in this analysis. These include N Reactor fuel, FFTF fuel, single-pass reactor (SPR) fuel, PWR Core-II fuel, and miscellaneous fuel. A sixth type of fuel, fuel wastes in EBR-II metal casks, was assumed to have similar shipping characteristics to miscellaneous fuels. Some of the key shipment characteristics for these fuels are presented in Table 5.11-1, including the SNF material forms, quantities, shipment capacities, and numbers of shipments. Radionuclide inventories for the various types of fuel shipments are provided in Table 5.11-2. The radionuclide inventories were derived from the irradiated fuel inventories and characteristics provided by Bergsman (1994, 1995) and the shipment characteristics listed in Table 5.11-2.

The population densities of the different areas of the Hanford Site across which shipments must travel will influence the transportation impacts. Doses to persons along the highways (i.e., off-link doses) will be received only by Hanford Site workers for onsite shipments.

Table 5.11-1. Spent nuclear fuel shipment characteristics.

Fuel Type	Material Form	Quantity, Assemblies	Shipment Capacity, Assemblies/shipment	Number of Shipments[a]
N Reactor	Uranium metal clad with Zircalloy-2	Short: 66,300 Long: 63,700	Short: 128 Long: 96	Short: 518 Long: 664 Total: 1,182
FFTF	Mixed uranium-plutonium oxide in stainless steel tubes	317	4	80
Single-pass reactor	Uranium metal enclosed in aluminum jackets	1,100	900	2
PWR Core-II	Natural uranium oxide clad in zirconium alloy	72	1	71
Fuel wastes in EBR-II metal casks	Plutonium-uranium compounds sealed in stainless steel canisters	24 casks	1 cask per shipment	24
Miscellaneous	Various uranium compounds from research and development programs	77	4	20

a. This column provides the number of onsite shipments projected to occur in the Decentralization, 1992/1993 Planning Basis, Regionalization, and Centralization Alternatives. For the No-Action Alternative, one shipment of N Reactor fuel currently at PUREX and all of the miscellaneous fuels were assumed to be transported onsite.

Table 5.11-2. Radionuclide inventories for shipments of each type of spent nuclear fuel on the Hanford Site (Ci/shipment).[a,b]

Radio-nuclide	FFTF	N Reactor	PWR Core-II fuel	Single-pass reactor	EBR-II/ Misc.[c]
H-3	2.1E+02	3.9E+03	1.6E+02	3.9E+03	0.0E+00
Mn-54	7.0E+02	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Fe-55	6.9E+02	1.1E+03	6.1E+03	1.1E+03	0.0E+00
Co-60	7.3E+02	7.9E+02	4.2E+03	7.9E+02	4.3E+02
Ni-63	6.0E+01	0.0E+00	2.7E+03	0.0E+00	0.0E+00
Kr-85	1.8E+03	7.5E+04	1.6E+03	7.5E+04	6.3E+02
Sr-90	1.3E+04	8.7E+05	1.8E+04	8.7E+05	3.1E+02
Y-90	1.3E+04	8.7E+05	1.8E+04	8.7E+05	3.1E+02
Ru-106	1.8E+04	7.1E+03	2.9E+02	7.1E+03	1.4E+03
Rh-106	1.8E+04	7.1E+03	2.9E+02	7.1E+03	1.4E+03
Sb-125	3.7E+03	1.6E+04	1.1E+03	1.6E+04	0.0E+00
Te-125m	9.1E+02	4.3E+03	2.6E+02	4.3E+03	0.0E+00
Cs-134	5.2E+03	1.9E+04	1.6E+03	1.9E+04	0.0E+00
Cs-137	3.6E+04	1.1E+06	3.6E+04	1.1E+06	3.5E+03
Ba-137m	3.4E+04	1.0E+06	3.4E+04	1.0E+06	3.3E+03
Ce-144	6.3E+03	4.1E+03	0.0E+00	4.1E+03	9.6E+03
Pr-144	6.3E+03	4.1E+03	0.0E+00	4.1E+03	9.6E+03
Pr-144m	7.6E+01	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Pm-147	2.8E+04	2.9E+05	4.5E+03	2.9E+05	7.7E+03
Sm-151	1.4E+03	1.3E+04	1.9E+02	1.3E+04	0.0E+00
Eu-154	1.0E+03	1.3E+03	2.1E+03	1.3E+03	0.0E+00
Eu-155	3.2E+03	4.8E+03	7.6E+02	4.8E+03	6.4E+01
U-233	0.0E+00	0.0E+00	0.0E+00	0.0E+00	1.3E-01
U-234	0.0E+00	1.5E+00	0.0E+00	1.5E+00	2.1E+01
U-235	2.0E-04	6.7E-02	0.0E+00	6.7E-02	2.6E-02
U-238	2.7E-02	1.0E+00	0.0E+00	1.0E+00	3.3E-04
Np-237	4.6E-02	3.5E-02	0.0E+00	3.5E-02	0.0E+00
Pu-238	6.6E+02	0.0E+00	1.1E+03	0.0E+00	3.8E+01
Pu-239	1.4E+03	1.8E+02	2.8E+02	1.8E+02	6.9E+01
Pu-240	1.5E+03	4.5E+01	3.7E+02	4.5E+01	2.0E+02
Pu-241	6.3E+04	1.7E+03	6.8E+04	1.7E+03	1.1E+04
Pu-242	5.2E-01	3.0E-03	0.0E+00	3.0E-03	6.9E-01
Am-241	8.0E+02	3.1E+01	1.6E+03	3.1E+01	0.0E+00
Cm-243	4.6E+01	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Cm-244	8.8E+01	0.0E+00	7.9E+02	0.0E+00	0.0E+00

a. Radionuclide inventory data were derived from information in Bergsman (1994) and WHC (1993c).
b. For radionuclides that are indicated to have 0.0 Ci per shipment, the quantities of fission and activation are less than 5 Ci/assembly and less than 10 g/assembly for actinides. Radionuclides not listed on the table are also less than these quantities.
c. Fuel inventories for EBR-II casks are assumed to be applicable to miscellaneous fuels. The SNF in EBR-II casks and miscellaneous SNF consist primarily of irradiated light-water reactor fuels.

The population densities for each work area on the site, used for occupational dose calculations, are listed in Table 5.11-3. The off-link doses are included in the occupational dose results.

For the calculation of doses to persons traveling on the highways (i.e., on-link doses), two-lane highways were assumed and the number of persons per vehicle was assumed to be 2.0. No vehicle stops were included in the calculations because the shipments are not long enough to warrant intermediate stops for food and rest. One-way traffic densities were based on traffic counts provided in DOE (1989). Because average traffic densities were not available in that document and there are no administrative restrictions on time of day when SNF transport could occur, the peak count on a given route segment (vehicles per day) was used to calculate the traffic density for that route. The traffic densities used for the five types of SNF and shipping distances for the various fuel types are provided below.

- FFTF Fuel - 640 vehicles per hour; 28 kilometers one-way shipping distance
- N Reactor Fuel - 170 vehicles per hour; 16 kilometers one-way shipping distance
- PWR Core II Fuel - 180 vehicles per hour; 5 kilometers one-way shipping distance
- Single-pass Reactor Fuel - 100 vehicles per hour; 16 kilometers one-way shipping distance
- EBR-II/300 Area Miscellaneous Fuel - 640 vehicles per hour; 37 kilometers one-way shipping distance.

Table 5.11-3. Population densities for work areas at Hanford.

Work Area	Worker Population	Land Area, km^2	Worker Density, per km^2
100 B and C	4	1.7	3
100 D and DR	4	1.5	3
100 H	4	0.7	6
100 K	124	0.9	140
100 N	360	1.0	360
200 West	1968	9.5	210
200 East	2923	9.0	330
300	2487	1.5	1700
400	638	2.1	300
600	514	1450	0.35
WPPSS	1125	4.4	260

The computer code RISKIND was used to calculate the doses to Maximally-Exposed Individual (MEI) members of the public as discussed in Volume 2, Chapter 5. Two exposure scenarios were modeled, including a "tailgater" and a "bystander." The dose received by a tail-gater was calculated by assuming that an individual precedes or follows an SNF shipment for the entire duration of a shipment. The exposure distance was assumed to be 48.8 meters (160 feet). The dose calculated in Volume 2, Chapter 5, was based on a 37 kilometers (23 miles) shipping distance, which is also the same as the longest shipping distance anticipated for SNF shipments at Hanford (300 Area to the 200 Area). Therefore, the public MEI dose amounts to 0.015 millirem per tailgating incident.

The dose to a "bystander" was calculated in Volume 2, Chapter 5, to be 0.0014 millirem. This dose was calculated assuming a shipment passes by an individual at an average speed of 56 kilometers per hour (35 miles per hour) at a distance of 1 meter (3 feet) from the shipment. This individual was postulated to be standing on the side of the road as an SNF shipment passes by and was assumed to be exposed only one time.

The dose to the maximally-exposed worker from incident-free transportation will be received by the truck crew. The dose to the truck crew was calculated using the maximum allowable dose rate in the truck cab (2 millirem per hour) for all shipments. It was assumed that the maximum-exposed worker will accompany all of the spent fuel shipments, even though the dose will most likely be apportioned over a larger number of workers. The total dose received by this individual was calculated by multiplying the maximum dose rate by the total shipping time. The total shipping time for the various alternatives was determined by dividing their total shipping distances by the average speed, 56 kilometers per hour (35 miles/hour).

The results of the analysis of the No Action Alternative are presented in Table 5.11-4. As shown, two shipment campaigns occur in this alternative; 1) shipment of N Reactor fuels at PUREX to the 105-K basins for storage and 2) shipment of miscellaneous SNF in the 300 Area to the 200 Area to be placed in dry storage. The total radiological impacts from incident-free transportation in this alternative are dominated by the shipments of miscellaneous fuels from the 300 Area to the 200 Area. This is primarily because there are approximately 24 shipments of miscellaneous fuels, and the N Reactor fuel at PUREX will make up only a fraction of a shipment.

Table 5.11-4. Impacts of incident-free transportation for the No Action Alternative.[a]

Impacts[b]	General Population[c]	Occupational
Total Dose (person-rem)	7.8E-02	1.2E-01
Cancer Fatalities	3.9E-05	4.7E-05

a. The N Reactor fuel currently at PUREX is the only N Reactor fuel transported in this alternative. The impacts of transporting this fuel were calculated by adjusting the impacts of transporting all N Reactor fuel (0.3 MTHM at PUREX/2096 MTHM total N Reactor fuel).
b. Total detriment, which includes latent cancer fatalities, nonfatal cancers, and genetic effects in subsequent generations, can be calculated by multiplying the total dose to the general population by 7.3E-04 effects per person-rem and the total occupational dose by 5.6E-04 effects per person-rem.
c. Rural population density.

The doses to the maximally-exposed workers and members of the public are summarized below:

- The dose to a tailgater was calculated to be 0.015 millirem.
- The dose to a bystander was calculated to be 0.0014 millirem.
- The dose to a truck crewman that accompanies all of the spent fuel shipments in the No Action Alternative was calculated to be about 46 millirem.

The RISKIND computer code was used to calculate the radiological consequences of accidental releases of radioactive material during transportation. Consequences of severe, reasonably foreseeable accidents were calculated to workers and the offsite population. Workers were placed at a distance that maximizes the dose from a potential release. Hanford-specific population density data (see Beck et al. 1991) were used to assess the integrated doses to the offsite public, as described in Volume 2, Chapter 5.

As discussed in Appendix I, maximum radiological impacts were calculated for a severe, reasonably foreseeable accident. For this assessment, the consequences were assessed to populations and individuals assuming the most severe accident scenario with a probability greater than 1E-07. The methods and data described in Appendix I were used to calculate the accident probabilities of the various shipments in the No Action Alternative. Hanford-specific numbers of shipments and shipping distances were used in the calculations. Accident rate information from Saricks and Kvitek (1991) for urban areas in the State of Washington were used in the calculations. The results of these calculations indicate that the probabilities of the severe accident defined in Appendix I for the irradiated fuels transported in the No Action Alternative are less than the 1E-07 criteria. The most likely severe accident scenario was determined to be one involving shipments of miscellaneous fuels from the 300 Area. The probability of such an accident was calculated to be about 1E-09. As shown in Table 5.11-5, this is also the highest-consequence accident scenario for the No Action Alternative.

The impacts of potential severe transportation accidents for the No Action Alternative are shown in Table 5.11-5. The maximum exposed individual and public collective doses are shown in Table 5.11-5 for shipments of miscellaneous SNF in the 300 Area to dry storage in the 200 Area. This was determined to be the most severe reasonably foreseeable onsite transportation accident scenario for the No Action Alternative, even though its probability is significantly smaller than 1E-07, as discussed above. As shown, consequence estimates are presented for two atmospheric dispersion conditions; 1) neutral (Pasquill stability class D, wind speed = 4 meters per second) and 2) stable (Pasquill stability class F , wind speed = 1 meters per second).16

Table 5.11-5. Impacts of accidents during transportation for the No Action Alternative.[a]

	Dose Consequence		Cancer Fatalities		Point Estimate of Risk	
	Stability Category		Stability Category		Stability Category	
Exposure Group	D	F	D	F	D	F
Offsite Population[b]	1.4E+01 person-rem	1.1E+02 person-rem	6.8E-03	5.5E-02	6.8E-12	5.5E-11
Maximum Exposed Individual	5.0E-01 rem	1.7E+00 rem	2.0E-04	6.7E-04	2.0E-13	6.7E-13

a. The maximum-consequence onsite transportation accident for the No Action Alternative is one involving a shipment of miscellaneous fuels currently located in the 300 Area. This is also the most likely accident scenario, but its probability is below the 1E-07 criteria for a credible accident.
b. Rural population density.

Nonradiological impacts consist of fatalities that may result from traffic accidents as well as health effects from pollutants emitted from vehicles involved in onsite shipments of spent nuclear fuel. These risks are unrelated to the radioactive nature of the materials being transported. Nonradiological impacts from accidents were calculated using unit risk factors derived by Saricks and Kvitek (1991) that convey the estimated number of fatalities per unit distance traveled. The total nonradiological impacts are calculated by multiplying the total shipping distance traveled by onsite shipments by the appropriate unit risk factors.

The total nonradiological transportation impacts for the No Action Alternative were calculated to be less than one (1.9E-05) fatality.

5.11.2 Decentralization Alternative

Implications of implementing the Decentralization Alternative for interim storage of SNF on traffic and incident-free onsite transportation of SNF and materials supporting SNF storage are discussed in the following subsections.

5.11.2.1 Traffic. Under the Decentralization Alternative, the number of construction workers would range from about 220 to 870. During operations, the number of workers would range from about 1100 to 1300, depending on the option selected. This would add from 1 to 6 percent to the present workforce and to additional commuting traffic on the Hanford Site, assuming that the proportion of workers that take the bus to work or drive their own vehicles remains essentially constant.

5.11.2.2 Transportation. The same approaches and basic assumptions and data described in Section 5.11.1.2 for the No Action Alternative were used to assess the impacts of onsite transportation for the Decentralization Alternative. The key differences between the alternatives are the numbers of shipments and destinations. More SNF is transported in this alternative than in the No Action Alternative. In this alternative, all N Reactor SNF in the 105-K Basins is to be transported to the 200 Area for processing and/or storage, depending upon the particular suboption selected. The FFTF fuel is to be transported from the 400 Area to the 200 Area for storage. The PWR Core-II, single- pass reactor fuels, and 300 Area miscellaneous fuels are also to be transported to a new facility in the 200 Area for storage.

Table 5.11-6 presents the incident-free transportation impacts for the Decentralization Alternative. As shown in Table 5.11-6, the truck crews are the largest exposure group. The total doses were found to be dominated by the exposures received during transportation of N Reactor fuel. This is because there are significantly more truck shipments of N Reactor fuel in this alternative than shipments of other types of fuel.

The doses to the maximally-exposed workers and members of the public are summarized below:

- The dose to a tailgater was calculated to be 0.015 millirem.
- The dose to a bystander was calculated to be 0.0014 millirem.
- The dose to a truck crewman that accompanies all of the spent fuel shipments in the Decentralization Alternative was calculated to be about 800 millirem.

The worker MEI dose is higher than that calculated for the No Action Alternative because there are many more onsite spent fuel shipments in the Decentralization Alternative.

Table 5.11-7 presents the impacts of potential severe transportation accidents for the Decentralization Alternative. The maximum exposed individual and public collective doses are shown in Table 5.11-7 for two accident scenarios: the highest probability and highest consequence. As explained in the table footnotes, the probabilities of both scenarios are less than MEI 1E-07 criteria discussed in Appendix I. As shown, consequence estimates are presented for

Table 5.11-6. Impacts of incident-free transportation for the Decentralization Alternative.

Impacts[a]	General Population[b]	Occupational
Total Dose (person-rem)	4.3E-01	1.7E+00
Cancer Fatalities	2.2E-04	6.8E-04

a. Total detriment, which includes latent cancer fatalities, non-fatal cancers, and genetic effects in subsequent generations, can be calculated by multiplying the total dose to the general population by 7.3E-04 effects per person-rem and the total occupational dose by 5.6E-04 effects per person-rem.
b. Rural population density.

Table 5.11-7. Impacts of accidents during transportation for the Decentralization Alternative.

Accident Scenario	Exposure Group	Dose Consequence		Cancer Fatalities		Point Estimate of Risk	
		Stability Category		Stability Category		Stability Category	
		D	F	D	F	D	F
Highest Probability[a]	Offsite Population[b]	1.7E+01 Person-rem	1.4E+02 Person-rem	8.6E-03	6.8E-02	4.3E-10	3.4E-09
	Maximum Exposed Individual	7.2E-01 Rem	2.4E+00 Rem	2.9E-04	9.6E-04	1.4E-11	4.8E-11
Highest Consequence[c]	Offsite Population	1.7E+02 Person-rem	1.3E+03 Person-rem	8.4E-02	6.7E-01	5.0E-10	4.0E-09
	Maximum Exposed Individual	5.4E+00 Rem	1.8E+01 Rem	2.2E-03	7.2E-03	1.3E-11	4.3E-11

a. The highest-probability accident is one involving a shipment of N Reactor fuel. The probability of this accident scenario was calculated to be approximately 5E-8 over the entire N-Reactor fuel shipping campaign.
b. Rural population density.
c. The highest-consequence accident scenario was determined to be one involving shipments of FFTF fuel. However, the probability of the accident scenario analyzed here is approximately 6E-09, which is below the 1E-07 probability criteria for a reasonably foreseeable accident.

two atmospheric dispersion conditions; 1) neutral (Pasquill stability class D, wind speed = 4 meters per second) and 2) stable (Pasquill stability class F , wind speed = 1 meters per second). This table is different from Table 5.11-5 (No Action Alternative) because of the additional fuel types transported in the Decentralization Alternative.

The total nonradiological transportation impacts for the Decentralization Alternative were calculated to be 6.6E-04 fatalities. The nonradiological transportation impacts of this alternative are significantly higher than the impacts of the No Action Alternative because the numbers of shipments, and thus total shipment mileage, is significantly higher.

5.11.3 1992/1993 Planning Basis Alternative

Implications of implementing the 1992/1993 Planning Basis Alternative for interim storage of SNF on traffic and incident-free onsite transportation of SNF and materials supporting SNF storage are discussed in the following subsections.

5.11.3.1 Traffic. Because the only difference between the Decentralization Alternative and the 1992/1993 Planning Basis Alternative is the shipment of the small amount of TRIGA fuel offsite, traffic patterns would not be significantly different from those described for the Decentralization Alternative.

5.11.3.2 Transportation. The impacts of onsite transportation for the 1992/1993 Planning Basis Alternative are substantially the same as the impacts of the Decentralization Alternative (see Section 5.11.2). The only difference between these two alternatives is the disposition of the TRIGA fuel in the 308 Building. The quantity and number of TRIGA fuel shipments is small relative to the other fuel types so the disposition of the TRIGA fuels will have a negligible impact on the results presented in Tables 5.11-3 and 5.11-4.

5.11.4 Regionalization Alternative

Implications of implementing the Regionalization Alternative for interim storage of SNF on traffic and incident-free onsite transportation of SNF and materials supporting SNF storage are presented in this section. The onsite transportation requirements for the four Regionalization Alternative options are as follows:

- Option A - Defense production fuel will be shipped from the 105-K basins and Plutonium and Uranium Recovery through Extraction to a new facility in the 200 Area for storage. All other fuel will be shipped offsite; the transportation impacts of offsite shipments are addressed in Appendix I.
- Option B1 - All SNF located or to be generated west of the Mississippi River will be sent to Hanford for storage, except for Naval SNF. Shipments of SNF from offsite locations are addressed in Appendix I. The onsite SNF will be transported from its current locations to the 200 Area for storage. In terms of onsite transportation impacts, this option is essentially the same as the Decentralization Alternative (see Section 5.11.2).

- Option B2 - The same as Option B1 except that Naval SNF will also be transported to Hanford. This alternative would result in the same onsite transportation impacts as Option B1.

- Option C - All Hanford SNF will be transported offsite to a facility at INEL or NTS. Offsite transportation impacts are addressed in Appendix I.

5.11.4.1 Traffic. Under the Regionalization Option A, the number of construction workers would range from about 180 to 1200, depending on the option selected. During operations, the number of workers would range from about 280 to 320, depending on the suboption selected. This would add from less than 1 to about 5 percent to the present work-force and to additional commuting traffic on the Hanford Site, assuming that the proportion of workers that take the bus to work or drive their own vehicles remains essentially constant. Assuming that all of the N Reactor fuel shipments travel 16 kilometers (10 miles) one way (approximate distance from the 100 Areas to the 200 Area), a total of about 40,000 vehicle-kilometers are needed for the N Reactor fuel shipments in this option. It was stated in Section 4.11 that in 1988 DOE vehicles logged over 19,000,000 vehicle-kilometers (12,000,000 vehicle-miles) at Hanford. The increase in vehicle mileage resulting from the Regionalization Option A, assuming that all the Hanford SNF shipments will be made in one year, is less than 1 percent above the 1988 base DOE-vehicle mileage.

For the Regionalization options B1 and B2, the impacts on traffic would be essentially the same as those described for the Decentralization Alternative (see Section 5.11.2.1).

The Regionalization Option C involves offsite shipments of Hanford fuel. The number of Hanford workers would stay approximately the same as the No Action Alternative. The impacts on traffic are predominantly related to the additional vehicles on the highways that are carrying Hanford fuels to INEL or NTS. Assuming that all of the onsite Hanford fuel shipments travel 48 kilometers (30 miles) one way (approximate distance from the 100 Areas to the 300 Area), a total of about 130,000 vehicle-miles are needed for the onsite segments of these shipments. It was stated in Section 4.11 that in 1988 DOE vehicles logged over 12,000,000 miles at Hanford. The increase in vehicle mileage resulting from Regionalization Option C, assuming that all the Hanford fuel shipments will be made in one year, is about 1 percent above the 1988 base DOE-vehicle mileage.

5.11.4.2 Transportation. In Regionalization Option A, all N Reactor SNF in the 105-K basins and at PUREX would be transported to the 200 Area for processing and/or storage, depending on the particular suboption selected. The FFTF, PWR Core-II, single-pass reactor fuels, and 300 Area miscellaneous fuels are to be transported to INEL. Offsite transportation impacts are addressed in Appendix I. Onsite transportation impacts for this option, therefore, would consist of the impacts of transporting N Reactor fuel from the 105-K basins and PUREX to the 200 Area.

The transportation impacts of this option were calculated by determining the impacts of transporting N Reactor fuel on a per-shipment basis and then multiplying the total number of shipments. The methods and input data described in Section 5.11.1 were used to calculate the per-shipment impacts. The results of the transportation impact calculations for the Regionalization Option A are as follows:

- Incident-free transportation impacts: Public exposures - 2.4E-01 person-rem (9.6E-05 LCFs); Worker exposures - 1.4E+00 person-rem (5.6E-04 LCFs).
- Impacts of transportation accidents: Public, Pasquill Stability Class D - 1.7E+01 person-rem (8.6E-03 LCFs); Public - Pasquill Stability Class F - 1.4E+02 person-rem (6.8E-02 LCFs). Maximum exposed individual, Pasquill Stability Class D - 7.2E-01 rem (2.9E-04 LCFs); Maximum exposed individual Pasquill Stability Class F - 2.9E+00 rem (9.6E-04 LCFs). See the "highest probability" accident in Table 5.11-7.
- Nonradiological impacts: 5.6E-04 fatalities.

The incident-free doses to the maximally-exposed workers and members of the public are summarized below:

- The dose to a tailgater was calculated to be 0.015 millirem.
- The dose to a bystander was calculated to be 0.0014 millirem.
- The dose to a truck crewman who accompanies all of the SNF shipments in Regionalization Option A was calculated to be about 680 millirem.

The worker MEI dose is higher than that calculated for the No Action Alternative because there are many more onsite spent fuel shipments in the Regionalization Option A.

The worker MEI dose is lower than that calculated for the Decentralization Alternative because only N Reactor fuel is shipped onsite in Regionalization Option A, and all fuel types are shipped onsite in the Decentralization Alternative.

In Regionalization options B1 and B2, all Hanford SNF would be shipped onsite from its current locations to the 200 Area. Traffic and transportation impacts for both Regionalization options B1 and B2 would be essentially the same as those calculated for the Decentralization Alternative.

In Regionalization Option C, all of the Hanford Site SNF would be shipped to and stored at either INEL or NTS. Because all of the shipments of Hanford SNF would be considered to be offsite shipments, the impacts are addressed in Appendix I. For Hanford, this option is identical to the Centralization Alternative, minimum option.

5.11.5 Centralization Alternative

Implications of implementing the Centralization Alternative for interim storage of SNF on traffic and incident-free onsite transportation of SNF and materials supporting SNF storage are discussed in the following subsections.

5.11.5.1 Traffic. Traffic patterns would be essentially the same as for the Decentralization Alternative if Hanford were selected to receive all DOE SNF. The patterns would last for up to twice as long because of the additional fuel to be brought to the reprocessing/ stabilization and storage facility (although there is only 25 weight percent more fuel to be shipped, it would likely require smaller quantities per shipment because of its higher heat load). If all Hanford fuel were to be shipped offsite, traffic patterns would not be significantly different from those of the No Action Alternative.

5.11.5.2 Transportation. The Centralization Alternative results in the same onsite transportation impacts as the Decentralization Alternative. In the Decentralization Alternative, all Hanford Site SNF will be transported to the 200 Areas for further processing and/or storage, depending on the specific option. In the Centralization Alternative, all Hanford Site SNF is transported to either a stabilization/packaging facility in the 200 Area for preparation for offsite shipment or to the Central Storage Facility to be located in the 200 Area. All of these cases requires onsite shipment of Hanford SNF from their current locations to a 200 Area facility.

Therefore, the onsite transportation impacts for the Centralization Alternative are the same as those for the Decentralization Alternative (see Section 5.11.2).

5.12 Occupational and Public Health and Safety

Implications of implementing the alternatives for interim storage of SNF on worker and public health and safety at the Hanford Site are discussed in the following subsections. By and large this material consists of summary material extracted from Section 5.7, "Air Quality and Related Consequences;" 5.8, "Water Quality and Related Consequences;" 5.11, "Traffic and Transportation;" and 5.15, "Accidents."

5.12.1 No Action Alternative

Radiological and nonradiological consequences relating to occupational and public health and safety for the No Action Alternative are presented in the following subsections.

5.12.1.1 Radiological Consequences. The consequences of air emissions from routine operations of existing facilities utilized in the No Action Alternative include a maximum annual dose of 1E-5 rem to a potential onsite worker with a 5E-9 probability of fatal cancer. The collective annual dose to workers in spent fuel storage facilities is 24 person-rem per year (Bergsman 1995), which would require about 60 years of such operation to accumulate a collective worker dose from which one fatal cancer might be inferred.

The dose to an offsite resident at the highest exposure location is estimated as 3E-6 rem/year, and the corresponding probability of fatal cancer is 1E-9.

The peak collective dose to the population within 80 kilometers (50 miles) is 3E-2 person-rem per year, which is predicted to result in less than one fatal cancer (about 36,000 years of such operation would be required to reach a dose from which one fatal cancer might be inferred).

5.12.2 Decentralization Alternative

Radiological and nonradiological consequences relating to occupational and public health and safety for the Decentralization Alternative are presented in the following subsections.

5.12.2.1 Radiological Consequences. The consequences of air emissions from individual facilities in the Decentralization Alternative are summarized in Table 5.7-8 and include a maximum annual dose of 2E-9 rem to a potential onsite worker (8E-13 probability of fatal cancer) for any combination of wet or dry spent fuel storage facilities. The dose to an offsite resident at the highest exposure location is estimated as 6E-10 rem per year, and the corresponding probability of fatal cancer is 3E-13. The peak collective dose to the population within 80 km is 2E-5 person-rem per year, which is predicted to result in less than one fatal cancer. The collective annual dose to workers at SNF facilities for a combination of wet and dry storage facilities is 2 person-rem per year for maintenance and operations. Loading the new facilities would require an additional 17-18 person-rem depending on the form of dry storage. For dry storage only, the dose from initial loading would be 7-12 person-rem, and there would be no dose from normal operations (Bergsman 1995).

For dry storage of defense fuel, stabilization prior to dry storage is included in the routine operations of the Decentralization Alternative, and additional emissions would result from these activities. The dose to the onsite worker from air emissions would increase by 4E-6 rem/year for a shear/leach/calcine process or 3E-5 rem/year for a solvent extraction process (2E-9 or 1E-8 probability of fatal cancer, respectively). Collective worker dose at fuel stabilization facilities would range from 44 person-rem per year at a shear/leach/calcine facility to 78 person-rem per year at a solvent extraction facility over the 4 years in which these facilities are expected to operate (Bergsman 1995). The dose to an individual worker in the facility is assumed to be limited by administrative controls to no more than 0.5 rem per year.

The consequences from stabilization for the offsite resident would be 7E-6 rem per year (4E-9 probability of fatal cancer) for the shear/leach/calcine facility and 2E-5 rem per year (1E-8 probability of fatal cancer) for the solvent extraction facility. The collective dose to the offsite population from the respective fuel stabilization facilities is estimated at 0.3 to 1 person-rem per year, resulting in less than one fatal cancer (would require from about 1000 to 3700 years of such exposure to reach a dose from which one fatal cancer might be inferred).

5.12.3 1992/1993 Planning Basis Alternative

Because the activities are similar, radiological consequences of routine operations for the 1992/1993 Planning Basis Alternative are considered to be the same as those for the Decentralization Alternative.

5.12.4 Regionalization Alternative

Radiological and nonradiological consequences relating to occupational and public health and safety for the Regionalization Alternative are presented in the following subsections.

5.12.4.1 Radiological Consequences. Because of the similarity of activities, the radiological consequences of routine operations for the Regionalization Alternative Option A are considered to be the same as those for the Decentralization Alternative. The consequences to the public of options B and C are the same as described in the following section for the Centralization Maximum and Minimum options, respectively. Consequences to onsite workers would differ based on the processing and storage options for onsite fuel as in the decentralization alternative, as well as on the quantity of imported fuel to be received and placed into dry storage under each option. The consequences over the 40-year storage period range from 98 to 320 person-rem for option A, 700-920 person-rem for options B1 and B2, and 190-320 person-rem for option C. No fatal cancers would be expected as a result of implementing any of these options.

5.12.5 Centralization Alternative

Radiological and nonradiological consequences relating to occupational and public health and safety for the Centralization Alternative are presented in the following subsections.

5.12.5.1. Radiological consequences of air emissions from routine operations in the Centralization Alternative include a maximum annual dose of 9E-9 rem to a potential onsite worker (4E-12 probability of fatal cancer) for any combination of wet or dry spent fuel storage facilities. The collective annual dose to SNF facility workers for a combination of wet and dry storage facilities is 2 person-rem per year for maintenance and operations. Loading the new facilities would require an additional 19-22 person-rem depending on the form of dry storage. For dry storage only, the dose from initial loading would be 9-12 person-rem, and there would be no dose from normal operations (Bergsman 1995). Shear/leach/calcine and solvent extraction activities would add 44 or 78 person-rem per year, respectively, and the receiving, canning, and technology development facilities would entail an additional 20 person-rem per year.

The dose from air emissions to an offsite resident at the highest exposure location is estimated as 2E-9 rem per year, and the corresponding probability of fatal cancer is 8E-13. The peak collective dose to the population within 80 kilometers (50 miles) is 7E-5 person-rem per year, which is predicted to result in less than one fatal cancer. These estimates do not include relocation of the expended core facility to Hanford, which is discussed in Appendix D to Volume 1 of this EIS. Assumptions used in the Appendix D calculations for consequences of locating an expended core facility at Hanford may differ from those used for other Hanford facilities.

5.13 Site Services

Implications of implementing the alternatives for interim storage of SNF on site services at the Hanford Site are discussed in the following subsections.

5.13.1 No Action Alternative

Implementing the No Action Alternative would require no significant additional consumption of material or energy; however, about 12,000 megawatt-hours per year are currently used for SNF management activities.

5.13.2 Decentralization Alternative

Incremental requirements for materials and energy in construction associated with the Decentralization Alternative are shown in Table 5.13-1. Annual consumption of energy during operations is similar to that used during construction for the water storage options (W and X), the total would be a small fraction of the present consumption rate. Annual consumption of energy during operations in the options where defense production fuel is stabilized is significantly greater; however it is still within the capacity of existing facilities.

Table 5-13-1. Materials and energy required for Decentralization suboptions.

Item	Option W	X	Y	Z	P	Q
Concrete, thousand cubic meters/(cubic yards)	13 (17)	15 (20)	17 (23)	24 (32)	22 (29)	29 (38)
Carbon steel, thousand tonnes (tons)	2.4 (2.7)	2.8 (3.1)	3.3 (3.6)	4.5 (5.0)	3.9 (4.2)	5.1 (5.6)
Stainless steel, thousand tonnes (tons)	0.1 (0.1)	0.1 (0.1)	0	0	0.5 (0.6)	0.7 (0.8)
Copper, thousand tonnes (tons)	0	0	0	0	0.06 (0.07)	0.08 (0.09)
Lumber, thousand cubic meters (board feet)	1.2 (500)	1.4 (570)	1.6 (650)	2.2 (930)	2.0 (850)	2.6 (1100)
Asphalt, sand, and crushed rock, thousand cubic meters (thousand cubic yards)	0.6 (0.8)	0.7 (0.9)	0.8 (1.1)	1.2 (1.5)	1.1 (1.4)	1.4 (1.8)
Electricity						
Construction (MW-hrs)	2500	2900	3500	4800	4370	5700
Operations (MW-hrs/yr)	1600	1600	100	100	40,000[a]	127,000[a]
Diesel fuel, thousand cubic meters (thousand gallons)	0.5 (130)	0.6 (150)	0.7 (175)	0.9 (240)	0.8 (220)	1.1 (290)
Gasoline, thousand cubic meters (thousand gallons)	0.5 (130)	0.6 (150)	0.7 (175)	0.9 (240)	0.8 (220)	1.1 (290)
Construction Cost ($ Million)	265	280	350	310	580	835

a. Assumes operation of the process facility (28,000 or 115,000 MW-hrs/yr) concurrently with those facilities where SNF is currently stored (12,000 MW-hrs/yr, as in the No Action Alternative) for an interim period less than 4 years.

In the Decentralization Alternative, an extension of existing utilities to the project site area would likely be necessary. This would include water mains, electrical power lines, sewage facilities, telephone lines, etc. All of these utilities are available in the adjacent 200-East Area. In addition, an existing rail line might need to be upgraded for increased traffic, and construction of new spurs going to various proposed new facilities would likely be required. The project would be served by an 8-inch water main capable of delivering 7600 liters per minute (2000 gallons per minute). Facilities would be designed to preclude discharge of water except for sanitary waste.

5.13.3 1992/1993 Planning Basis Alternative

Energy requirements in the 1992/1993 Planning Basis Alternative would be essentially the same as those cited above for the Decentralization Alternative.

5.13.4 Regionalization Alternative

Material and energy requirements in the Regionalization Option A would be slightly less than those cited above for the Decentralization Alternative. Material and energy requirements in the Regionalization options would be similar to those cited above for the Decentralization Alternative, although the construction requirements would occur over most of the interim storage period. Incremental requirements for materials and energy in construction associated with the Regionalization options are shown in Tables 5.13-2 and 5.13-3. For the Regionalization options that involve fuel from other locations being stored at the Hanford Site, the requirements shown are for fuel received from other locations and are in addition to those shown in Table 5.13-1 for fuel already at the Hanford Site. For the Regionalization option that has no fuel stored at the Hanford Site, the requirements shown are the total incremental requirements.

5.13.5 Centralization Alternative

Similar to the Decentralization Alternative, annual consumption of energy during operations is similar to that used during construction for the water storage options (W and X), and the total would be a small fraction of the present consumption rate. Annual consumption of energy during operations in the options where defense production fuel is stabilized is significantly greater; however it is still within the capacity of existing facilities. Materials and energy requirements for construction in the Centralization Alternatives are shown in Table 5.13-4. Similar to the Regionalization options, the Centralization Alternative that involves fuel from other locations being stored at the Hanford Site shows the requirements associated with storing the fuel received from other locations and are in addition to those shown for fuel already at the Hanford Site in Table 5.13-1. For the Centralization option that has no fuel stored at the Hanford Site, the requirements shown are the total incremental requirements.

In the Centralization Alternative where all SNF is brought to the Hanford Site, an extension of existing utilities to the project site area would be necessary. This would include water mains, electrical power lines, sewage facilities, telephone lines, etc. All of these utilities

Table 5-13-2. Materials and energy required for Regionalization A suboptions.

Item	Option W	X	Y	Z	P	Q
Concrete, thousand cubic meters/(cubic yards)	9 (12)	9 (12)	16 (21)	19 (25)	22 (29)	29 (38)
Carbon steel, thousand tonnes (tons)	1.7 (1.9)	1.7 (1.9)	3.0 (3.4)	3.6 (4)	3.9 (4.2)	5.1 (5.6)
Stainless steel, thousand tonnes (tons)	0.1 (0.1)	0.1 (0.1)	0	0	0.5 (0.6)	0.7 (0.8)
Copper, thousand tonnes (tons)	0	0	0	0	0.06 (0.07)	0.08 (0.09)
Lumber, thousand cubic meters (board feet)	0.8 (350)	0.8 (350)	1.4 (600)	1.7 (700)	2.0 (850)	2.6 (1100)
Asphalt, sand, and crushed rock, thousand cubic meters (thousand cubic yards)	0.5 (0.6)	0.5 (0.6)	0.8 (1.0)	0.9 (1.2)	1.1 (1.4)	1.4 (1.8)
Electricity						
Construction (MW-hrs)	1800	1800	3200	3800	4370	5700
Operations (MW-hrs/yr)	1600	1600	100	100	40,000[a]	127,000[a]
Diesel fuel, thousands cubic meters (thousand gallons)	0.4 (100)	0.4 (100)	0.6 (160)	0.7 (190)	0.8 (220)	1.1 (290)
Gasoline, thousand cubic meters (thousand gallons)	0.4 (100)	0.4 (100)	0.6 (160)	0.7 (190)	0.8 (220)	1.1 (290)
Construction Cost ($ Million)	200	200	340	250	580	835

a. Assumes operation of the process facility (28,000 or 115,000 MW-Hrs/yr) concurrently with those facilities where SNF is currently stored (12,000 MW-Hrs/yr, as in the No Action Alternative) for an interim period less than 4 years.

Table 5-13-3. Materials and energy required for construction of Regionalization B and C options.

Item	SNF Stored at the Hanford Site Without Naval SNF	SNF Stored at the Hanford Site With Naval SNF	No SNF Stored at the Hanford Site
	Option		
Concrete, thousand cubic meters/(cubic yards)	54 (70)	115 (150)	18 (23)
Carbon steel, thousand tonnes (tons)	8.2 (9)	19.1 (21)	3.1 (3.4)
Stainless steel thousand tonnes (tons)	0.1 (0.1)	0.1 (0.1)	0.4 (.5)
Copper, thousand tonnes (tons)	0	0	0.05 (0.05)
Lumber, thousand cubic meters (board feet)	4.8 (2000)	10 (4200)	1.6 (660)
Asphalt, sand, and crushed rock, thousand cubic meters (thousand cubic yards)	2.5 (3.3)	5.4 (7.1)	0.8 (1.1)
Electricity			
Construction (MW-hrs)	16,000	30,000	3400
Operations (MW-hrs/yr)[a]	100-127,000	100-127,000	0-20,000
Diesel fuel, thousand cubic meters (thousand gallons)	1.9 (500)	4.2 (1100)	0.6 (170)
Gasoline, thousand cubic meters (thousand gallons)	1.9 (500)	4.2 (1100)	0.6 (170)
Construction Cost ($ Million)	765	1465	560

a. Minimum value represents requirements during the period after all fuel has been placed into dry storage, or has been shipped offsite. Maximum value represents requirements during the interim period (less than 4 years) while SNF is being processed and prepared for storage or shipment offsite, assuming concurrent operation of the process facility and the existing facilities where SNF is currently stored (as in the No Action Alternative).

are available in the adjacent 200-East Area. In addition, an existing rail line might need to be upgraded for increased traffic and the construction of new spurs to various proposed new facilities would likely be required.

The following section describes the material requirements for operation of facilities in each SNF alternative and the corresponding quantities of waste generated by these activities. Table 5.14-1 lists the breakdown by alternative and suboption of the various types of waste generated by SNF management facilities.

Table 5-13-4. Materials and energy requirements for construction of Centralization options.

Item	No Fuel Stored at the Hanford Site	All Offsite Fuel Stored at the Hanford Site
Concrete, thousand cubic meters (cubic yards)	18 (23)	150 (200)
Carbon Steel, thousand tonnes (tons)	3.1 (3.4)	25 (27.5)
Stainless Steel, thousand tonnes (tons)	0.4 (0.5)	0.1 (0.1)
Copper, thousand tonnes (tons)	0.045 (0.05)	0
Lumber, thousand cubic meters (board feet)	1.6 (660)	13 (5600)
Asphalt, Sand, and Crushed Rock (thousand cubic meters (thousand cubic yards)	0.8 (1.1)	7.2 (9.5)
Electricity		
Construction (MW-hrs)	3400	40,000
Operations (MW-hrs/yr)[a]	0-20,000	100-127,000
Diesel fuel, thousand cubic meters (thousand gallons)	0.6 (170)	5.7 (1500)
Gasoline, thousand cubic meters (thousand gallons)	0.6 (170)	5.7 (1500)
Construction Cost ($ Million)	560	1950

a. Minimum value represents requirements during the period after all fuel has been placed into dry storage, or has been shipped offsite. Maximum value represents requirements during the interim period (less than 4 years) while SNF is being processed and prepared for storage or shipment offsite, assuming concurrent operation of the process facility and the existing facilities where SNF is currently stored (as in the No Action Alternative).

5.14 Materials and Waste Management

5.14.1 No Action Alternative

The No Action Alternative involves only fuel storage at existing facilities, and material requirements for the current configuration are minimal. The exception is make-up water for the 105-K fuel storage basins, which amounts to 2.8 million cubic meters per year.

The quantity of waste generated in the No Action Alternative is also relatively small because the only planned modifications to existing facilities are safety and security upgrades to the 105-K basins. About 530 cubic meters of low-level waste would result from containerization of SNF in 105-KE Basin, and small quantities of radioactive and mixed waste are generated at the 325 Building.

Table 5.14-1. Waste generation for spent nuclear fuel management alternatives.

		Decentralization						Centralization	
Waste Type	No Action	W	X	Y	Z	P	Q	Offsite	at Hanford [a,b]
Construction Waste (m^3, total)	0	1500	1700	1700	2800	2600	3400	2000	15000
High-Level Radioactive Waste (m^3/y)	0	0	0	0	0	0	57	14	0
Transuranic Waste (m^3/y)	0	0	0	0	0	28	50	0	0
Low-Level Radioactive Waste (m^3/y)[c]	95	41	50	0	0	280	420	140	68
Mixed Waste (Low-Level Radioactive and Hazardous, (m^3/y)	0.96	0.23	0.23	0	0	2.0	2.0	1.0	0.28
Non-radioactive Hazardous Waste (m^3/y)	2.3	1.1	1.1	0	0	2.8	2.8	1.4	1.1

a. These quantities are associated with new facilities that would be required for management of SNF shipped to Hanford from other sites. They represent incremental increases over those for facilities that are required to manage SNF currently at Hanford, which are discussed in the No-Action and Decentralization Alternatives.

b. A new ECF is not included in these totals; requirements for this facility are discussed in Volume 1, Appendix D.

c. Annual totals do not include containerization of defense production reactor SNF currently stored at the 105-K basins. This activity is expected to generate 530 cubic meters of low-level radioactive waste over a period of approximately 2 years.

Table 5.14-1. (contd)

Waste Type	Regionalization AX	AY	AZ	AP	AQ	B1[a]	B2[a,b]	C
Construction Waste (m^3, total)	900	1600	2100	2600	3400	5400	11,500	2000
High-Level Radioactive Waste (m^3/y)	0	0	0	0	57	0	0	14
Transuranic Waste (m^3/y)	0	0	0	28	50	0	0	0
Low-Level Radioactive Waste (m^3/y)[c]	61	0	0	280	420	1.7	1.7	140
Mixed Waste (Low-Level Radioactive and Hazardous, (m^3/y)	0.23	0	0	2.0	2.0	0.028	0.028	1.0
Non-radioactive Hazardous Waste (m^3/y)	1.1	0	0	2.8	2.8	0.057	0.057	1.4

a. These quantities are associated with new facilities that would be required for management of SNF shipped to Hanford from other sites. They represent incremental increases over those for facilities that are required to manage SNF currently at Hanford, which are discussed in the No-Action and Decentralization Alternatives.

b. A new ECF is not included in these totals; requirements for this facility are discussed in Volume 1, Appendix D of this document.

c. Annual totals do not include containerization of defense production reactor SNF currently stored at the 105-K basins. This activity is expected to generate 530 cubic meters of low-level radioactive waste over a period of approximately 2 years.

5.14.2 Decentralization Alternative

Material requirements for the Decentralization Alternative depend on the suboption chosen. The suboptions involving wet storage of production reactor fuel (suboptions W and X) require make-up water for the storage basin at approximately 2300 cubic meters per year. Material requirements for dry storage of fuel (suboptions Y and Z) are minimal, and consist of decontamination chemicals in small quantities. Those suboptions including processing of production reactor fuel (suboptions P and Q, which would be combined with either Y or Z) require relatively large quantities of nitric acid (2000 - 4000 cubic meters per year) and other process chemicals in smaller quantities.

Construction waste generated for each of the suboptions depends on the size and number of facilities required. Dry storage of all fuel, including processing of production reactor fuel, would result in the largest quantity of construction waste, which is assumed to be nonradioactive, nonhazardous solids. Radioactive and hazardous waste from operations is also greater for the dry storage suboption with processing. Wet storage of production reactor fuel and dry storage of other onsite fuel results in the smallest quantity of both construction and operational hazardous waste.

5.14.3 1992/1993 Planning Basis Alternative

This alternative would be essentially the same as the Decentralization Alternative at Hanford.

5.14.4 Regionalization Alternative

Regionalization Alternative Option A would be essentially the same as the Decentralization Alternative at Hanford in terms of operational material requirements and waste generation because these originate largely from the storage pool or process facilities, depending on the suboption selected. The quantity of construction waste would be smaller because the dry storage capacity for nondefense production fuel would not be needed.

The Regionalization Alternative B options would require materials in similar quantities to the Decentralization Alternative, but would generate construction and operational wastes in greater quantities because of additional facilities that would be necessary to receive, package, and store imported SNF. Note that the waste quantities reported in Table 5.14-1 represent incremental increases for SNF facilities above those listed for the Decentralization Alternative.

The Regionalization Alternative Option C involves only stabilization of defense production fuel and packaging of all Hanford SNF for shipment offsite. It is identical to the Centralization Alternative minimum option as described in Section 5.14.5.

5.14.5 Centralization Alternative

The Centralization Alternative minimum option for offsite shipment of Hanford fuel requires construction of a stabilization and canning facility, which would produce annual quantities of construction and operational wastes similar to those for onsite combined wet and dry storage (suboptions W and X) in the Decentralization Alternative. However, these wastes would only be generated for the time required to stabilize and package fuel for offsite shipment (approximately 4 years).

Centralization at Hanford (maximum option) would include the same suboptions as Decentralization for SNF currently at Hanford, and the material requirements and waste generation would be identical. For SNF imported from other sites, additional dry storage capacity would be needed, and new additional facilities to package and examine the fuel would be constructed. The estimates in Table 5.14-1 for Centralization at Hanford represent incremental increases for these additional facilities above those in the Decentralization Alternative. They do not incorporate the additional requirements of the Expended Core Facility, which are discussed in Volume 1, Appendix D of this document. Operational material requirements for the incremental dry storage capacity would be minimal, as would be the quantities of waste generated. Construction of the new facilities would generate nonhazardous solid waste in quantities greater than any of the other options, but operation of the additional facilities would produce relatively small quantities of radioactive and hazardous waste.

5.15 Facility Accidents

Implications of facility accidents associated with implementing the alternatives for SNF storage at Hanford are discussed in the following section. The method used to screen and select accidents for analysis is described, as are the procedures for evaluating the consequences of selected accidents, and the results of the analysis. Additional detail concerning specific accidents and parameters used in the analysis is provided in Attachment A, Facility Accidents.

5.15.1 Historical Accidents Involving SNF at Hanford

There are no known instances at Hanford where storage, handling, or processing of SNF has resulted in an accident that involved a significant release of radioactive or other hazardous materials to the environment or that resulted in detrimental exposure of workers or members of the public to hazardous materials.

5.15.2 Emergency Preparedness Planning at Hanford

Although the safety record for operations at Hanford and other DOE facilities is generally good, DOE-RL and all Hanford Site contractors have established Emergency Response Plans to prepare for and mitigate the consequences of potential emergencies on the Hanford Site (DOE 1992c). These plans were prepared in accordance with DOE Orders and other federal, state, and local regulations. The plans describe actions that will be taken to evaluate the severity of a potential emergency and the steps necessary to notify and coordinate the activities of other agencies having emergency response functions in the surrounding communities. They also specify levels at which the hazard to workers and the public are of sufficient concern that protective action should be taken. The Site holds regularly scheduled exercises to ensure that individuals with responsibilities in emergency planning are properly trained in the procedures that have been implemented to mitigate the consequences of potential accidents and other events.

5.15.3 Accident Screening and Selection for the EIS Analysis

The alternatives for SNF storage considered in this EIS necessitate evaluation of accidents at a variety of different types of facilities. In the No Action Alternative, the facilities consist of those where SNF is currently stored on the Hanford Site, or those where SNF will be stored at the time of the record of decision. All facilities considered in the No Action Alternative currently exist at the Hanford Site, and no construction of new facilities is assumed. For many of these facilities, storage of SNF is incidental to other activities that take place in the buildings. For the other alternatives (Decentralization, Regionalization, 1992/1993 Planning Basis, and Centralization), construction of new facilities dedicated solely to SNF management is assumed.

Accidents evaluated for existing facilities at Hanford consisted of maximum reasonably foreseeable accidents described in such previously published analyses as safety or NEPA documentation. The source documents for specific accidents evaluated in this section are referenced in the detailed accident descriptions in Attachment A. In the case of new facilities, hypothetical accidents were based on operation of similar facilities at Hanford or other sites. Depending on the time at which the source document was prepared, the number and types of accidents considered for each facility would be somewhat variable. However, the screening process used in the relatively recent analyses considers a wide scope of accident initiators and scenarios, including industrial accidents (fires, explosions, overpressurization, loss of containment or confinement), criticality, operator error or injury, external hazards (surface vehicle or aircraft impact), waste management, natural phenomena (seismic events, wind, floods, volcanic activity), interactions with activities at adjacent facilities (construction, maintenance, operations), and common cause events (power failure). Older safety documents generally address these issues as well, although perhaps not with the same rigor as newer analyses. Transportation accidents are considered in a separate section of this appendix and are not discussed here.

Acts of terrorism are accounted for indirectly in the present analysis because the potential consequences of terrorist activities are used to determine security requirements for a given facility. Security measures are implemented to mitigate the impact, or reduce the probability, of high consequence events. Therefore, reasonably foreseeable scenarios for terrorist activities would entail risks that are similar to those for the types of accident initiators generally considered in the source documents that provide the basis for this analysis.

For the purposes of this EIS, accidents are ideally grouped into three categories based on their estimated frequencies as follows: abnormal events (frequency $\geq 10^{-3}$ per year), design basis accidents (frequencies $<10^{-3}$ to 10^{-6} per year), and beyond design basis accidents (frequency $<10^{-6}$ to 10^{-7} per year). Because the accident categories commonly used for development of safety documents encompass different probability ranges, the estimated frequencies (or frequency ranges) for Hanford facility accidents are reported as indicated in the source document without regard to the accident frequency categories established for use in the EIS. For accidents where only a range rather than a point estimate of frequency is available, the frequency of the accident is reported as being less than the highest frequency that defines the range. In alternatives that consider SNF imported from other sites (such as other DOE facilities or U.S. and foreign research reactors), frequencies for specific accidents have been adjusted to account for increased fuel handling at receiving, canning, and storage facilities.

Accident frequencies as reported in safety documents (Safety Analysis Reports and related analyses) typically represent the overall probability of the accident, including the probability of the initiating event combined with the frequency of any contributing events required for an environmental release to occur. The contributing events may include equipment or barrier failures, or failures of other mitigating systems designed to prevent accidental releases. In general, the safety documents do not evaluate the consequences of events with expected frequencies of $< 10^{-6}$ per year because such accidents are not considered reasonably foreseeable; therefore, accidents in the beyond design basis category are generally not evaluated for this analysis. Evaluation of aircraft traffic at the Richland and Pasco, Washington airports determined that impacts of commercial or military aircraft were less than $1x10^{-7}$ for a facility in the Hanford 300 Area, which is at highest risk because of its location (PNL 1992a). Therefore, aircraft accidents are not considered further in this analysis as initiators for accidents at Hanford SNF management facilities.

As noted previously, the safety documents for SNF facilities generally considered a broad range of accidents; however, only the consequences of the maximum reasonably foreseeable accidents for each facility in a given alternative were evaluated for this document. Of the existing facilities assessed in the No Action Alternative, most are multipurpose facilities with diverse missions such as research or process development. These facilities typically contain relatively small quantities of SNF relative to the 105-K basins, where the bulk of Hanford's existing SNF is stored. The accidents evaluated in the source documents for multipurpose facilities may therefore reflect activities other than SNF storage or handling. The risks for such accidents are reported in this EIS for completeness, although in some cases, neither the frequency nor the consequences associated with the accident depend on the presence of SNF in the facility.

5.15.4 Method for Accident Consequence Analysis

In the No Action Alternative, accident consequence analyses utilized release estimates as presented in the source document for a given existing facility. For new facilities, release estimates were based on historical operation of similar facilities at Hanford. These estimates were also assumed to represent typical accidental releases in alternatives that consider storage of fuel from offsite locations, such as other DOE facilities or U.S. and foreign research reactors. Accidents evaluated for the research reactor fuels indicate that releases for such specialized fuels would be comparable to those included in this analysis (DOE 1993b; Hale and Reutzel 1993). The assumptions used to determine radionuclide releases are included in Attachment A.

Because most source documents (other than the more recent Safety Analysis Reports) do not evaluate hazardous materials other than radionuclides, a different approach was used for accidents involving nonradioactive materials. The hazardous material inventories for each facility were used to estimate releases based on the physical state of each compound as described in Attachment A. Specific initiators and accident scenarios were generally not postulated for nonradioactive materials; therefore, frequencies were not estimated for hazardous chemical accidents.

The downwind concentrations for materials released in accidents were then calculated at receptor locations as defined for the EIS. The receptors included a worker who is onsite but outside the facility where the accident takes place, a member of the public who is temporarily at the nearest access location (such as a road that crosses the site or at the site boundary), and the maximally exposed offsite resident. Collective dose to the population within 80 kilometers (50 miles) was also calculated for radionuclide releases. Individual dispersion calculations were performed using 95 percent atmospheric conditions (those resulting in air concentrations that would not be exceeded more than 5 percent of the time). Dose to the population was calculated using both 50 percent and 95 percent atmospheric dispersion parameters. Dispersion calculations were performed using the GENII computer code (Napier et al. 1988) for radionuclide releases and the EPIcode (Homann 1988) for nonradioactive compounds.

The radiation dose to each receptor evaluated for the EIS was recalculated for the specific conditions and release location as appropriate to each alternative using the GENII computer code. Doses were calculated as the effective dose equivalent using standard assumptions for the Hanford Site as summarized in Schreckhise et al. (1993). Health effects were also estimated as probability of fatal cancer based on recommendations of the International Commission on Radiological Protection in its Publication 60 (ICRP 1991). The accident doses were recalculated for this analysis using a consistent, reasonably conservative set of methods and assumptions and to include the complete set of receptors that are to be evaluated in the EIS. This was necessary because the methods used in the source documents were not necessarily consistent and in some cases were outdated. For this reason, the doses developed for this analysis may differ from those reported in the source documents that describe the accidents; however, they should be viewed as a screening analysis for the purposes of the EIS and are not intended to replace or invalidate the previous results.

Individual doses were based on exposure of the receptor during the entire release, except where the release time was sufficiently long that such an assumption is unrealistic. For releases that were expected to last more than a few hours, the exposure duration for onsite workers and members of the public at accessible onsite locations was limited to 2 hours, corresponding to the maximum time required to evacuate the Hanford Site in the event of an accident. Offsite residents were assumed to be exposed during the entire release, regardless of the accident duration. Exposure via inhalation and external pathways (groundshine and submersion in the plume) were considered for workers and the nearest public access receptors; ingestion of contaminated food was evaluated only for offsite residents. Because protective action guidelines specify mitigative actions to prevent consumption of contaminated food, the ingestion dose to offsite individuals and populations is reported separately from the other exposure routes. Reduced exposure to the plume or to contaminated ground surface as a result of early evacuation of offsite populations is not assumed for the purposes of this analysis, although such actions would also be mandated if the projected dose from an accident exceeded the protective action guidelines. Because the circumstances and consequences postulated for workers at the scene of an accident are so speculative, they serve no useful purpose in the decision-making process. As a consequence, discussion of impacts on "close-in" workers are not brought forward into the text of this Appendix. Consequences in terms of the "close-in" workers for one scenario in each accident may be found in Attachment A.

5.15.5 Radiological Accident Analysis

5.15.5.1 No Action Alternative. The No Action Alternative consists of fuel storage at existing Hanford facilities, including the 100-K wet storage basins; T Plant, and a low-level burial ground in the 200-West Area; the 308, 324, 325, and 327 buildings in the 300 Area; and the Fast Flux Test Facility (FFTF) in the 400 Area. Of these facilities, only the 100-K storage basins and the FFTF fuel storage facility are primarily devoted to SNF storage; the others are all multipurpose facilities that house a variety of activities in addition to storing relatively small quantities of SNF. The consequences and risks of accidents associated with these facilities are described in Tables 5.15-1 through 5.15-5.

The maximum reasonably foreseeable accident for multipurpose facilities is an earthquake scenario at the 324 Building, which releases non-SNF related radioactive material that has accumulated in a hot cell (Table 5.15-1 through Table 5.15-5). The contributions of other activities at the facility, including SNF storage, are estimated to be relatively minor. The maximum reasonably foreseeable accident directly involving SNF management is a fire at a fuel storage facility adjacent to FFTF. Several of the accident scenarios evaluated for this alternative involve initiators that could affect more than one facility (e.g., earthquakes); however, the combined consequences of releases from potentially affected facilities have not been evaluated for a common receptor.

5.15.5.2 Decentralization Alternative. The Decentralization Alternative involves several options for construction of new facilities at Hanford. One option includes a combination of new wet storage for defense production reactor fuel currently stored at the 105-K basins and new dry storage for fuel that is currently at other locations. Alternative options are included for processing of production reactor fuel prior to dry storage. The consequences of accidents at the new facilities are based on previously evaluated accidents for similar installations, adapted for the conditions and location of these facilities as assumed in this EIS.

The maximum reasonably foreseeable accident for the new facilities is a severe cask impact followed by a fire at a dry storage facility (Tables 5.15-1 through 5.15-5). The risk from a cask drop while loading fuel at a wet storage facility is similar for most receptors, although this scenario is conservative for a new facility as discussed in Attachment A.

5.15.5.3 1992/1993 Planning Basis Alternative. Accidents and consequences would be essentially the same as for the Decentralization Alternative.

5.15.5.4 Regionalization Alternative. The consequences of the regionalization alternatives are similar to those of other action alternatives because they only differ in the quantity of imported fuel placed into dry storage at the site. The types of facilities and activities involved are generally the same as those considered for the decentralization and centralization alternatives. Point estimates of risk for some accidents differ from those of corresponding

Table 5.15-1. Radiological accidents, individual worker probability of latent cancer fatality.

Accident Description	Attribute	No Action	Decentralization	1992/1993 Planning Basis	Regionalization A, B	Centralization at Hanford	Regionalization or Centralization - Other Site
SNF facilities:							
Wet storage fuel cask drop	Consequences	1.4E-03	3.5E-04	3.5E-04	3.5E-04	3.5E-04	NA[a]
	Annual Frequency	<1E-04	<1E-04	<1E-04	<1E-04	<1E-04	NA
	Point Estimate of Risk	<1.4E-07	<3.5E-08	<3.5E-08	<3.5E-08	<3.5E-08	NA
FFTF liquid metal fire in fuel storage	Consequences	2.4E-07	NA	NA	NA	NA	NA
	Annual Frequency	<1E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	<2.9E-11	NA	NA	NA	NA	NA
Multi-Purpose Facilities:							
324 Building Seismic event[c]	Consequences	(b)	NA	NA	NA	NA	NA
	Annual Frequency	4E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	(b)	NA	NA	NA	NA	NA
325 Building Seismic event	Consequences	1.0E-01	NA	NA	NA	NA	NA
	Annual Frequency	2E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	2.0E-05	NA	NA	NA	NA	NA
308 Building Fuel transfer accident	Consequences	5.2E-06	NA	NA	5.2E-06	NA	NA
	Annual Frequency	<1E-02	NA	NA	<1E-02	NA	NA
	Point Estimate of Risk	<5.2E-08	NA	NA	<5.2E-08	NA	NA

Table 5.15-1. (contd)

Accident Description	Attribute	No Action	Decentralization	1992/1993 Planning Basis	Regionalization A	Regionalization B	Centralization at Hanford	Regionalization or Centralization - Other Site
New dry storage - cask impact & fire	Consequences	NA[a]	9.4E-02	9.4E-02	9.4E-02	9.4E-02	9.4E-02	9.4E-02
	Annual Frequency	NA	6E-06	6E-06	6E-06	7E-06	8E-06	5E-06
	Point Estimate of Risk	NA	5.6E-07	5.6E-07	5.6E-07	6.6E-07	7.5E-07	4.7E-07
New SNF process - U metal fire	Consequences	NA	8.3E-08	8.3E-08	8.3E-08	8.3E-08	8.3E-08	8.3E-08
	Annual Frequency	NA	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04
	Point Estimate of Risk	NA	<8.3E-12	<8.3E-12	<8.3E-12	<8.3E-12	<8.3E-12	<8.3E-12
New ECF	Consequences	NA	NA	NA	NA	(c)	(c)	NA
	Annual Frequency	NA	NA	NA	NA	—[d]	—	NA
	Point Estimate of Risk	NA	NA	NA	NA	—	—	NA

a. NA = Not applicable.
b. The dose from this scenario (1.1E + 03) rem is sufficiently high that application of a fatal cancer risk factor is inappropriate.
c. See Appendix D for consequences of accidents at this facility.
d. Dash indicates that the information was not available.
e. The consequences associated with this accident are a result of existing contamination in the 324 Building hot cells, and neither its likelihood nor its severity depend on the presence of spent nuclear fuel at the facility. The actual contribution of spent nuclear fuel to releases from the accident is assumed to be negligible compared with that of other sources.

Table 5.15-2. Radiological accidents, general population - 80 km latent cancer fatalities, 95% meteorology.

Accident Description	Attribute	No Action	Decentralization	1992/1993 Planning Basis	Regionalization A, B	Centralization at Hanford	Regionalization or Centralization - Other Site
SNF Facilities:							
Wet Storage Fuel Cask Drop	Consequences	6.9E+00	3.0E+00	3.0E+00	3.0E+00	3.0E+00	NA[a]
	Annual Frequency	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04	NA
	Point Estimate of Risk	<6.9E-04	<3.0E-04	<3.0E-04	<3.0E-04	<3.0E-04	NA
FFTF Liquid Metal Fire in Fuel Storage	Consequences	3.2E+01	NA	NA	NA	NA	NA
	Annual Frequency	<1.0E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	<3.2E-03	NA	NA	NA	NA	NA
Multipurpose Facilities:							
324 Building Seismic Event[c]	Consequences	9.7E+02	NA	NA	NA	NA	NA
	Annual Frequency	4E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	3.9E-01	NA	NA	NA	NA	NA
325 Building Seismic Event	Consequences	2.0E+00	NA	NA	NA	NA	NA
	Annual Frequency	2E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	4.0E-04	NA	NA	NA	NA	NA
308 Building Fuel Transfer Accident	Consequences	NE[b]	NA	NA	NE	NA	NA
	Annual Frequency	<1.0E-02	NA	NA	—[c]	NA	NA
	Point Estimate of Risk	—	NA	NA	—	NA	NA

Table 5.15-2. (contd)

Accident Description	Attribute	No Action	Decentralization	1992/1993 Planning Basis	Regionalization A	Regionalization B	Centralization at Hanford	Regionalization or Centralization - Other Site
New dry storage - cask impact & fire	Consequences	NA	8.1E+01	8.1E+01	8.1E+01	8.1E+01	8.1E+01	8.1E+01
	Annual Frequency	NA	6E-06	6E-06	6E-06	7E-06	8E-06	5E-06
	Point Estimate of Risk	NA	4.9E-04	4.9E-04	4.9E-04	5.7E-04	6.5E-04	4.1E-04
New SNF process - U metal fire	Consequences	NA	6.4E-02	6.4E-02	6.4E-02	—[c]	6.4E-02	6.4E-02
	Annual Frequency	NA	<1.0E-04	<1.0E-04	<1.0E-04	—	<1.0E-04	<1.0E-04
	Point Estimate of Risk	NA	<6.4E-06	<6.4E-06	<6.4E-06	—	<6.4E-06	<6.4E-06
New ECF	Consequences	NA	NA	NA	NA	—	(d)	NA
	Annual Frequency	NA	NA	NA	NA	—	—	NA
	Point Estimate of Risk	NA	NA	NA	NA	—	—	NA

a. NA = Not applicable.
b. NE = Collective dose not evaluated for this scenario.
c. Dash indicates that the information was not available.
d. See Appendix D for consequences.
e. The consequences associated with this accident are a result of existing contamination in the 324 Building hot cells, and neither its likelihood nor its severity depend on the presence of SNF at the facility. The actual contribution of SNF to releases from the accident is assumed to be negligible compared with that of other sources.

Table 5.15-3. Radiological accidents, general population - 80 km latent cancer fatalities, 50% meteorology.

Accident Description	Attribute	No Action	Decentralization	199219/93 Planning Basis	Regionalization A, B	Centralization at Hanford	Regionalization or Centralization - Other Site
SNF Facilities:							
Wet storage - fuel cask drop	Consequences	4.0E-01	1.9E-01	1.9E-01	1.9E-01	1.9E-01	NA[a]
	Annual Frequency	< 1.0E-04	< 1.0E-04	< 1.0E-04	< 1.0E-04	<1.0E-04	NA
	Point Estimate of Risk	<4.0E-05	< 1.9E-05	< 1.9E-05	< 1.9E-05	<1.9E-05	NA
FFTF liquid metal fire in fuel storage	Consequences	3.8E+00	NA	NA	NA	NA	NA
	Annual Frequency	< 1.0E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	< 3.8E-04	NA	NA	NA	NA	NA
Multipurpose Facilities:							
324 Building Seismic Event[c]	Consequences	1.0E+02	NA	NA	NA	NA	NA
	Annual Frequency	4E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	4.0E-02	NA	NA	NA	NA	NA
325 Building Seismic Event	Consequences	2.3E-01	NA	NA	NA	NA	NA
	Annual Frequency	2E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	4.6E-05	NA	NA	NA	NA	NA
308 Building fuel transfer accident	Consequences	NE[b]	NA	NA	NE	NA	NA
	Annual Frequency	<1.0E-02	NA	NA	—[c]	NA	NA
	Point Estimate of Risk	—	NA	NA	—	NA	NA

Table 5.15-3. (contd)

Accident Description	Attribute	No Action	Decentralization	1992/1993 Planning Basis	Regionalization A	Regionalization B	Centralization at Hanford	Regionalization or Centralization - Other Site
New dry storage - cask impact & fire	Consequences	NA	4.0	4.0	4.0	4.0	4.0	4.0
	Annual Frequency	NA	6E-06	6E-06	6E-06	7E-06	8E-06	5E-06
	Point Estimate of Risk	NA	2.4E-05	2.4E-05	2.4E-05	2.8E-05	3.2E-05	2.0E-05
New SNF process - U metal fire	Consequences	NA	4.6E-03	4.6E-03	4.6E-03	4.6E-03	4.6E-03	4.6E-03
	Annual Frequency	NA	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04
	Point Estimate of Risk	NA	<4.6E-07	<4.6E-07	<4.6E-07	<4.6E-07	<4.6E-07	<4.6E-07
New ECF	Consequences	NA	NA	NA	NA	(d)	(d)	NA
	Annual Frequency	NA	NA	NA	NA	—	—	NA
	Point Estimate of Risk	NA	NA	NA	NA	—	—	NA

a. NA = Not applicable.
b. NE = Collective dose not evaluated for this scenario.
c. Dash indicates that the information was not available.
d. See Appendix D for consequences of accidents at this facility.
e. The consequences associated with this accident are a result of existing contamination in the 324 Building hot cells, and neither its likelihood nor its severity depend on the presence of SNF at the facility. The actual contribution of SNF to releases from the accident is assumed to be negligible compared with that of other sources.

Table 5.15-4. Radiological accidents, nearest public access - individual probability of latent cancer fatality.

Accident Description	Attribute	No Action	Decentralization	1992/1993 Planning Basis	Regionalization A, B	Centralization at Hanford	Regionalization or Centralization - Other Site
SNF Facilities:							
Wet storage fuel cask drop	Consequences	1.3E-03	3.1E-05	3.1E-05	3.1E-05	3.1E-05	NA[a]
	Annual Frequency	<1E-04	<1E-04	<1E-04	<1E-04	<1E-04	NA
	Point Estimate of Risk	<1.3E-07	<3.1E-09	<3.1E-09	<3.1E-09	<3.1E-09	NA
FFTF liquid metal fire in fuel storage	Consequences	1.2E-07	NA	NA	NA	NA	NA
	Annual Frequency	<1E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	<1.2E-11	NA	NA	NA	NA	NA
Multipurpose facilities:							
324 Building Seismic Event[d]	Consequences	1.9E-01	NA	NA	NA	NA	NA
	Annual Frequency	4E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	7.6E-05	NA	NA	NA	NA	NA
325 Building seismic event	Consequences	6.3E-03	NA	NA	NA	NA	NA
	Annual Frequency	2E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	1.3E-06	NA	NA	NA	NA	NA
308 Building fuel transfer accident	Consequences	4.3E-07	NA	NA	4.3E-07	NA	NA
	Annual Frequency	<1E-02	NA	NA	<1E-02	NA	NA
	Point Estimate of Risk	<4.3E-09	NA	NA	<4.3E-09	NA	NA

Table 5.15-4. (contd)

Accident Description	Attribute	No Action	Decentralization	1992/1993 Planning Basis	Regionalization A	Regionalization B	Centralization at Hanford	Regionalization or Centralization - Other Site
New dry storage - cask impact and fire	Consequences	NA	3.8E-05	3.8E-05	3.8E-05	3.8E-05	3.8E-05	3.8E-05
	Annual Frequency	NA	6E-06	6E-06	6E-06	7E-06	8E-06	5E-06
	Point Estimate of Risk	NA	2.3E-10	2.3E-10	2.3E-10	2.7E-10	3.0E-10	1.9E-10
New SNF process - U metal fire	Consequences	NA	2.2E-08	2.2E-08	2.2E-08	2.2E-08	2.2E-08	2.2E-08
	Annual Frequency	NA	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04
	Point Estimate of Risk	NA	<2.2E-12	<2.2E-12	<2.2E-12	<2.2E-12	<2.2E-12	<2.2E-12
New ECF	Consequences	NA	NA	NA	NA	(c)	(c)	NA
	Annual Frequency	NA	NA	NA	NA	—	—	NA
	Point Estimate of Risk	NA	NA	NA	NA	—	—	NA

a. NA = Not applicable.
b. See Appendix D for consequences of accidents at this facility.
c. The consequences associated with this accident are a result of existing contamination in the 324 Building hot cells, and neither its likelihood nor its severity depend on the presence of SNF at the facility. The actual contribution of SNF to releases from the accident is assumed to be negligible compared with that of other sources.

Table 5.15-5. Maximum exposed offsite individual - probability of latent cancer fatality.

Accident Description	Attribute	No Action	Decentralization	1992/1993 Planning Basis	Regionalization A, B	Centralization at Hanford	Regionalization or Centralization - Other Site
SNF Facilities:							
Wet storage fuel cask drop	Consequences	2.5E-04[a]	1.8E-04	1.8E-04	1.8E-04	1.8E-04	NA[b]
	Annual Frequency	<1E-04	<1E-04	<1E-04	<1E-04	<1E-04	NA
	Point Estimate of Risk	<2.5E-08	<1.8E-08	<1.8E-08	<1.8E-08	<1.8E-08	NA
FFTF liquid metal Fire in fuel storage	Consequences	2.5E-04[a]	NA	NA	NA	NA	NA
	Annual Frequency	<1E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	2.5E-08	NA	NA	NA	NA	NA
Multipurpose Facilities:							
324 Building Seismic Event[d]	Consequences	2.5E-04[a]	NA	NA	NA	NA	NA
	Annual Frequency	4E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	1.0E-07	NA	NA	NA	NA	NA
325 Building Seismic Event	Consequences	2.5E-04[a]	NA	NA	NA	NA	NA
	Annual Frequency	2E-04	NA	NA	NA	NA	NA
	Point Estimate of Risk	5.0E-08	NA	NA	NA	NA	NA
308 Building fuel transfer accident	Consequences	4.3E-08	NA	NA	4.3E-08	NA	NA
	Annual Frequency	<1E-02	NA	NA	<1E-02	NA	NA
	Point Estimate of Risk	4.3E-10	NA	NA	4.3E-10	NA	NA

Table 5.15-5. (contd)

Accident Description	Attribute	No Action	Decentralization	1992/1993 Planning Basis	Regionalization A	Regionalization B	Centralization at Hanford	Regionalization or Centralization - Other Site
New dry storage - cask impact & fire	Consequences	NA	2.5E-04	2.5E-04	2.5E-04	2.5E-04	2.5E-04	2.5E-04
	Annual Frequency	NA	6E-06	6E-06	6E-06	7E-06	8E-06	5E-06
	Point Estimate of Risk	NA	1.5E-09	1.5E-09	1.5E-09	1.8E-09	2.0E-09	1.2E-09
New SNF process - U metal fire	Consequences	NA	3.4E-06	3.4E-06	3.4E-06	3.4E-06	3.4E-06	3.4E-06
	Annual Frequency	NA	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04	<1.0E-04
	Point Estimate of Risk	NA	<3.4E-10	<3.4E-10	<3.4E-10	<3.4E-10	<3.4E-10	<3.4E-10
New ECF	Consequences	NA	NA	NA	NA	(c)	(c)	NA
	Annual Frequency	NA	NA	NA	NA	—	—	NA
	Point Estimate of Risk	NA	NA	NA	NA	—	—	NA

a. The offsite dose from this accident is assumed to be limited to 0.5 rem by application of protective action guidelines. Potential dose without protective action is 1.4 rem for 105-K Basin Cask drop, 5400 rem for 324 Building seismic event, 16 rem for 325 Building seismic event, and 5 rem for FFTF liquid metal fire.

b. NA = Not applicable.

c. See Appendix D for consequences of accidents at this facility.

d. The consequences associated with this accident are a result of existing contamination in the 324 Building hot cells, and neither its likelihood nor its severity depend on the presence of SNF at the facility. The actual contribution of SNF to releases from the accident is assumed to be negligible compared with that of other sources.

accidents in the other alternatives because the frequencies were adjusted to account for the quantity of fuel handled in each option (See Tables 5.15-1 through 5.15-5). Under subalternatives A and B, the types of accidents and their consequences would be the same as those for the decentralization alternative. However, the frequencies (and therefore the risks), would differ in some cases because of the volume of imported fuel that would be placed into dry storage. For subalternative C, all fuel currently at Hanford would be transported to another site, and the risks would be identical to those in the centralization minimum alternative.

5.15.5.5 Centralization Alternative. The Centralization Alternative consists of two options at Hanford: a minimum option in which all DOE spent fuel at Hanford is transported offsite to another location for interim storage, and a maximum option that would result in storage of all DOE spent fuel at Hanford. Accident scenarios for the minimum option would include those discussed under the No Action Alternative prior to shipment of the fuel offsite. In addition, defense reactor fuel would be processed and repackaged in a new facility prior to shipment. The risks associated with this new facility are expected to be similar to the processing facility discussed under the Decentralization Alternative. The cask impact accident at a dry storage facility has been included in this option to account for handling of fuel prior to shipment from Hanford.

The maximum option contains suboptions for wet or dry fuel storage with processing similar to those for the Decentralization Alternative, and the consequences are expected to be essentially the same as those described previously. The frequency of the cask impact at a dry storage facility has been increased to account for additional fuel that would be handled at Hanford under this option. The only other installation that would be included in this option is the Expended Core Facility (ECF), which would be relocated from INEL. The consequences of accidents at this facility are discussed in Volume 1, Appendix D of this EIS, and are not described here. Note that the accident analysis for the ECF in Appendix D incorporates different assumptions than those used for other Hanford facilities in this section, and the two sets of results are not directly comparable. The consequences of ECF accidents at Hanford using assumptions consistent with those in this section would be higher than those reported in Appendix D.

5.15.6 Secondary Impacts of Radiological Accidents

Secondary impacts of radiological accidents have been evaluated qualitatively for this analysis. Accidents that resulted in doses to the maximally exposed offsite resident of less than 100 millirem were considered to have little or no secondary impact because the levels of environmental contamination in these cases would be relatively small. Accidents that exceed this level may have secondary impacts with severity depending on the expected levels of environmental contamination. Although the levels of environmental contamination were not assessed quantitatively for this analysis, the offsite individual dose provides a measure of the air concentration and radionuclide deposition at the receptor location and can be used as a semi-quantitative estimate of the level of environmental contamination from a given accident. The estimated secondary consequences of maximum reasonably foreseeable SNF facility accidents are presented in Table 5.15-6.

5.15.7 Nonradiological Accident Analysis

For purposes of the EIS, a worst case accident scenario was developed for each existing and planned facility. The details of the nonradiological accident scenario are presented in Attachment A, and the information is summarized in this section. The accident assumes that a chemical spill occurs within a building and is followed by an environmental release from the normal exhaust system. It is assumed that the building remains intact but containment measures fail, allowing releases occur through the ventilation system. It is assumed that all, or a portion of, the entire inventory of toxic chemicals stored in each building is spilled. The environmental releases are modeled, and the hypothetical concentrations at three receptor locations are compared to toxicological limits.

Several chemical inventory and chemical emissions lists are provided by alternative and facility (Bergsman 1995). Effects to onsite workers, the nearest point of public access, and the public at the nearest offsite residence were estimated using the computer model EPIcode (DOE 1993b). Results from the EPIcode model were compared to available Emergency Response Planning Guideline (ERPG) values, Immediately Dangerous to Life and Health (IDLH) values, and Threshold Limit Values/Time Weighted Averages (TLV/TWA). In the absence of these values, toxicological data for similar health endpoints, from the Registry of Toxic Effects for Chemical Substances (RTEC) are used.

The results of the accident scenario for each alternative are presented in Table 5.15-8. As a general statement, in the event of an accident, the existing 105-KE and 105-KW facilities and the proposed new wet storage facility present the predominant risk for chemical exposure.

Under the No Action Alternative there is a potential for irreversible health effects to occur in the 308, 324, 325 A and B buildings, while nitric acid is a potential odor and irritation problem from both of the proposed fuel stabilization alternatives.

5.15.7.1 No Action Alternative. A baseline of chemicals kept in spent nuclear storage facilities was developed from chemical inventories for these facilities compiled to comply with the Emergency Planning and Community Right-To-Know Act (EPCRA). The existing storage facilities include 105-KE, 105-KW, PUREX (202A), T-Plant (221T), 2736-ZB Building, 200-West low-level burial grounds, FFTF 403 Building, 308 Building, 324 Building, 325 A&B Building, and 327 Building. The Emergency Planning and Community Right-To-Know Act (EPCRA) lists used are from 1992.

Because most facilities have various missions, the need to have a supply of chemicals at these facilities may not be related to the storage of SNFs. However for purposes of the EIS, the assumption is made that the existing inventories represents the anticipated amounts and types of chemicals which may be needed in the future.

The results of the accident scenario under conditions of the No Action Alternative are presented in Table 5.15-7.

5.15.7.2 Decentralization Alternative. The Decentralization Alternative involves construction of several new facilities at Hanford, including new dry storage for spent fuel, or a combination of new wet and dry storage. Options are also included for several types of fuel processing prior to storage. The consequences of new facilities are based on previously evaluated accidents for similar installations, adapted for the conditions and locations of these facilities as assumed in this EIS.

The baseline chemical inventory for the proposed facilities is primarily derived from the facility costs section in the engineering design data (Bergsman 1995). However, the wet storage facility uses the 105-KE Basin as a surrogate for a baseline chemical inventory because the facility cost section lists only two chemicals, sodium hydroxide and sulfuric acid.

Table 5.15-6. Assessment of secondary impacts of accidents for the No-Action Alternative.

	Environmental or Social Factor							
Accident Description	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights, Cultural Resources, Native Cultures
Accidents with frequencies $\geq 10^{-3}$ per year								
308 Building (fuel handling accident)	a	a	a	a	a	a	a	a
Accidents with frequencies $< 10^{-3}$ per year								
324 Building (seismic event)	Potential local effects on individuals of some species	Potential temporary closure of Hanford Reach of Columbia River to boat traffic, restriction of water use locally (Richland, Pasco)	Possible loss of crops, cost incurred for clean-up	None anticipated	May be extensive in vicinity of facility and adjacent offsite areas	None anticipated	Restriction on use of adjacent land for agriculture, and of Columbia River islands, pending radiological survey	Possible temporary restrictions on access to traditional fishing sites
325 Building (seismic event)	b	b	b	b	b	b	b	b
FFTF fuel storage (liquid metal fire)	b	b	b	b	b	b	b	b
105-K wet storage (cask drop)	b	b	b	b	b	b	b	b
200-W burial ground (cask impact & fire)	b	b	b	b	b	b	b	b
327 Building (hot cell fire)	b	b	b	b	b	b	b	b
T-plant (fuel damage)	a	a	a	a	a	a	a	a

a. Consequences of this accident would be limited to very local onsite impact only, if any.

b. Consequences of this accident would be similar in nature to those of the 324 building or new dry storage facility (worst case) accidents; however they would be less severe because offsite concentrations would be lower by at least two orders of magnitude.

The results of the accident scenario under conditions of the Decentralization Alternative are presented in Table 5.15-8.

5.15.7.3 1992/93 Planning Basis Alternative. Accidents and consequences would be essentially the same as for the Decentralization Alternative.

5.15.7.4 Regionalization Alternative. Except for Regionalization Option C, which would be essentially the same as the Centralization Alternative minimum case, accidents and consequences for options A, B1, and B2 would be essentially the same as for the Decentralization Alternative. The quantity of nondefense fuels placed into dry storage would not affect the potential for releases of hazardous chemicals because no such materials are present in the dry storage facilities.

5.15.7.5 Centralization Onsite Alternative. The Centralization Onsite Alternative consists of consolidating all spent fuel at the Hanford site. Options are available for wet or dry fuel storage with processing similar to those for the Decentralization Alternative. The consequences are expected to be essentially the same as those described for the first 5 years of the No Action Alternative, and then they are the same as those described for the Decentralization Alternative.

The results of the accident scenario under conditions of the No Action and Decentralization Alternatives are presented in Table 5.15-8.

5.15.7.6 Centralization Offsite Alternative. The Centralization Offsite Alternative consists of transporting all DOE SNF at Hanford offsite to another location for interim storage. Fuel would be stabilized prior to shipment in a fuel drying and passivation facility. Therefore the impacts from this alternative are the same as those for the No Action Alternative for the first 5 years, and then they are the same as those described for the fuel drying and passivation facility.

The results of the accident scenario under conditions of the No Action Alternative and the fuel drying and passivation facility are presented in Table 5.15-8.

Table 5.15-7. Assessment of secondary impacts of accidents for the Decentralization, 1992/1993 Planning Basis, Regionalization, and Centralization Alternatives.

Accident Description	Environmental or Social Factor							
	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights/ Cultural Resources/ Native Cultures
New dry storage (cask impact with fire)	Minimal local effects	Possible temporary restriction of use of Columbia River for recreation	Clean-up costs locally, potential loss of crops	None antici-pated	Moderate in immediate environs & offsite	None antici-pated	Temporary restriction on agriculture pending radiological survey	Possible temporary restriction on access to traditional fishing sites
New process facility (U metal fire)	a	a	a	a	a	a	a	a
New wet storage (cask drop)	b	b	b	b	b	b	b	b

a. Consequences of this accident would be limited to very local onsite impact only, if any.

b. Consequences of this accident would be similar in nature to those of the 324 building or new dry storage facility (worst case) accidents; however they would be less severe because offsite concentrations would be lower by at least two orders of magnitude.

5.15.8 Construction and Occupational Accidents

Table 5.15-9 shows the predicted number of injuries, illnesses, and fatalities among workers from construction activities and operations activities for each alternative. Injury, illness, and fatality counts for construction workers are presented separately because of the relatively more hazardous nature of construction work.

Decentralization suboptions P and Q represent the highest predicted construction and occupational accident count of any of the alternatives. The higher number of accidents is attributable to increased construction and fuel processing required by these alternatives. The Centralization Onsite Alternative has accident counts similar to those for suboptions P and Q. The lowest accident counts are for the No Action Alternative and the Centralization Offsite Alternative. All other alternative are similar in their predicted accident counts.

5.16 Cumulative Impacts Including Past and Reasonably Foreseeable Actions

Cumulative impacts associated with implementing the alternatives for interim storage of SNF at the Hanford Site together with impacts from past and reasonably foreseeable future actions are described in the following subsections.

5.16.1 No Action Alternative

Cumulative impacts associated with implementation of the No Action Alternative are described in the following subsections.

5.16.1.1 Land Use. The Hanford Site consists of about 1450 square kilometers (360,000 acres), of which about 87 square kilometers (22,000 acres) have been disturbed. Implementation of the No Action Alternative would not change that land use. Construction of the Environmental Restoration Disposal Facility will require disturbance of approximately 4.1 square kilometers (1.020 acres) of land. However, restoration of existing disturbed sites will compensate for this loss.

Table 5.15-8. Nonradiological exposure to public and workers to chemicals in spent nuclear fuel storage locations released during an accident.

Alternative/ Facility/ Chemical	Worker Exposure mg/m3	Exposure at Nearest Public Access mg/m3	Exposure at Nearest Public Residence mg/m3	ERPG 1[a] or TLV/IWA mg/m3	ERPG 2[b] or 0.1 IDLH mg/m3	ERPG 3[c] or IDLH mg/m3
No Action						
105-KE						
chlorine	4.30	4.30	0.13	2.9[d]	8.7	58
PCB	23.00	23.00	0.66	0.5	0.5	5
sodium hydroxide	140.00	140.00	0.40	2	20	200
sulfuric acid	220.00	220.00	6.40	2	10	30
105-KW						
chlorine	4.30	4.30	0.13	2.9	8.7	58
ethylene glycol	2.40	2.40	0.07	127	300	3000
kerosene	15.00	0.86	0.43	100	500	5000
polyacrylamide	4.20	0.24	0.12	0.03	400	4000
sodium hydroxide	140.00	140.00	0.40	2	20	200
sulfuric acid	220.00	220.00	6.40	2	10	30
PUREX (202A)						
cadmium nitrate tetrahydrate	0.03	0.03	0.02	0.05	10.5	105
diesel fuel	1.80	1.70	1.10	7	170	1700
mercury	7.20E-04	6.90E-04	4.30E-04	0.01	1	10
methanol	2.10E-04	2.00E-04	1.30E-04	262	3276	32760
PCB	0.00	0.00	0.00	0.5	0.5	5
sodium hydroxide	0.03	0.03	0.01	2	20	200
sodium nitrite	0.04	0.04	0.03	96	960	9600
T-Plant (221T)						
potassium permanganate	0.01	0.00	0.00	2	10	30
sodium	0.10	0.01	0.00	2	20	200
sodium hydroxide	0.02	0.01	0.00	2	20	200
sodium nitrite	0.05	0.00	0.00	96	960	9600
FFTF (403 Building)						
sodium	67.00	24.00	0.83	2	20	200
sodium potassium alloy	5.40	2.70	0.39	2	20	200
308 Building						
acetone	0.03	0.02	0.01	1780	2000	20000
ethylene glycol	70.00	57.00	37.00	127	300	3000
x-ray film (Ag)	88.00	0.77	0.36	0.01	62	620

Table 5.15-8 (contd)

Alternative/ Facility/ Chemical	Worker Exposure mg/m3	Exposure at Nearest Public Access mg/m3	Exposure at Nearest Public Residence mg/m3	ERPG 1[a] or TLV/TWA mg/m3	ERPG 2[b] or 0.1 IDLH mg/m3	ERPG 3[c] or IDLH mg/m3
324 Bldg						
alkyl dimethyl benzyl ammonium	29.00	1.90	0.24	***10***	***13***	130
bis-tri-n-butyltin oxide	38.00	2.40	0.31	***0.1***	***20***	200
poly oedmi ethylene dichloride	82.00	5.20	0.68	***40***	400	4000
325 Building						
mercury	3.20	0.20	0.03	***0.01***	***1***	10
poly oedmi ethylene dichloride	21.00	1.30	0.17	40	400	4000
zinc	0.04	0.00	0.00	5	12.4	124
327 Building						
poly oedmi ethylene dichloride	0.05	0.01	0.04	40	400	4000
Decentralization Suboption W						
Wet Storage Facility						
chlorine	0.75	0.10	0.04	2.9	8.7	58
PCB	3.90	0.54	0.20	***0.5***	***0.5***	5
sodium hydroxide	36.00	1.10	0.06	***2***	***20***	200
sulfuric acid	39.00	5.30	2.00	***2***	***10***	30
Vault Dry Storage Facility						
no chemicals of concern						
Decentralization Suboption X						
Wet Storage Facility						
chlorine	0.75	0.10	0.04	2.9	8.7	58
PCB	3.90	0.54	0.20	***0.5***	***0.5***	5
sodium hydroxide	36.00	1.10	0.06	***2***	***20***	200
sulfuric acid	39.00	5.30	2.00	***2***	***10***	***30***
Casks Dry Storage Facility						
no chemicals of concern						
Decentralization Suboption Y						
Vault Dry Storage Facility						
no chemicals of concern						
Shear\Leach\Calcine Stabilization Facility						
diesel fuel	0.42	0.40	0.26	7	170	1700
nitric acid	21.00	20.00	13.00	***2***	25.8	258
sodium hydroxide	0.86	0.73	0.20	2	20	200
sodium nitrite	0.11	0.10	0.06	96	960	9600
sulfuric acid	0.53	0.51	0.32	2	10	30

Table 5.15-8 (contd)

Alternative/ Facility/ Chemical	Worker Exposure mg/m3	Exposure at Nearest Public Access mg/m3	Exposure at Nearest Public Residence mg/m3	ERPG 1[a] or TLV/TWA mg/m3	ERPG 2[b] or 0.1 IDLH mg/m3	ERPG 3[c] or IDLH mg/m3
Decentralization Suboption Z						
Casks Dry Storage Facility						
no chemicals of concern						
Shear\Leach\Calcine Stabilization Facility						
diesel fuel	0.42	0.40	0.26	7	170	1700
nitric acid	21.00	20.00	13.00	***2***	25.8	258
sodium hydroxide	0.86	0.73	0.20	2	20	200
sodium nitrite	0.11	0.10	0.06	96	960	9600
sulfuric acid	0.53	0.51	0.32	2	10	30
Decentralization Suboption P						
105-KE						
chlorine	4.30	4.30	0.13	***2.9***	8.7	58
PCB	23.00	23.00	0.66	***0.5***	***0.5***	***5***
sodium hydroxide	140.00	140.00	0.40	***2***	***20***	***200***
sulfuric acid	220.00	220.00	6.40	***2***	***10***	***30***
105-KW						
chlorine	4.30	4.30	0.13	***2.9***	8.7	58
ethylene glycol	2.40	2.40	0.07	127	300	3000
kerosene	15.00	0.86	0.43	100	500	5000
polyacrylamide	4.20	0.24	0.12	***0.03***	400	4000
sodium hydroxide	140.00	140.00	0.40	***2***	***20***	200
sulfuric acid	220.00	220.00	6.40	***2***	***10***	***30***
Shear\Leach\Calcine Stabilization Facility						
diesel fuel	0.42	0.40	0.26	7	170	1700
nitric acid	21.00	20.00	13.00	***2***	25.8	258
sodium hydroxide	0.86	0.73	0.20	2	20	200
sodium nitrite	0.11	0.10	0.06	96	960	9600
sulfuric acid	0.53	0.51	0.32	2	10	30
Decentralization Suboption Q						
105-KE						
chlorine	4.30	4.30	0.13	***2.9***	8.7	58
PCB	23.00	23.00	0.66	***0.5***	***0.5***	***5***
sodium hydroxide	140.00	140.00	0.40	***2***	***20***	***200***
sulfuric acid	220.00	220.00	6.40	***2***	***10***	***30***

Table 5.15-8 (contd)

Alternative/ Facility/ Chemical	Worker Exposure mg/m3	Exposure at Nearest Public Access mg/m3	Exposure at Nearest Public Residence mg/m3	ERPG 1[a] or TLV/TWA mg/m3	ERPG 2[b] or 0.1 IDLH mg/m3	ERPG 3[c] or IDLH mg/m3
105-KW						
chlorine	4.30	4.30	0.13	***2.9***	8.7	58
ethylene glycol	2.40	2.40	0.07	127	300	3000
kerosene	15.00	0.86	0.43	100	500	5000
polyacrylamide	4.20	0.24	0.12	***0.03***	400	4000
sodium hydroxide	140.00	140.00	0.40	***2***	***20***	200
sulfuric acid	220.00	220.00	6.40	***2***	***10***	***30***
Solvent Extraction Fuel Stabilization Facility						
cadmium nitrate tetrahydrate	0.03	0.03	0.02	0.05	10.5	105
diesel fuel	0.42	0.40	0.26	7	170	1700
hydrazine	0.02	0.02	0.01	0.13	10.5	104.8
kerosene	0.84	0.81	0.51	100	500	5000
nitric acid	21.00	20.00	13.00	***5.2***	25.8	258
potassium permanganate	0.00	0.00	0.00	2	10	30
sodium hydroxide	0.86	0.73	0.20	2	20	200
sodium nitrite	0.11	0.10	0.06	96	960	9600
sulfuric acid	0.53	0.51	0.32	2	10	30
1992/1993 Planning Basis						
same as Decentralization						
Regionalization						
same as Decentralization						
Centralization Onsite						
same as No Action for first 5 years, then						
same as Decentralization						
Centralization Offsite						
same as No Action for first 5 years, then						
same as fuel drying and passivation facility						
Fuel Drying and Passivation Facility						
diesel fuel	0.42	0.40	0.26	7	170	1700

Table 5.15-8 (contd)

Alternative/ Facility/ Chemical	Worker Exposure mg/m3	Exposure at Nearest Public Access mg/m3	Exposure at Nearest Public Residence mg/m3	ERPG 1[a] or TLV/TWA mg/m3	ERPG 2[b] or 0.1 IDLH mg/m3	ERPG 3[c] or IDLH mg/m3
sodium hydroxide	0.09	0.07	0.02	2	20	200
sodium nitrite	0.11	0.10	0.06	96	960	9600
sulfuric acid	0.53	0.51	0.32	2	10	30

a. Emergency Response Planning Guideline (ERPG) value 1 (irritation or odor), or Threshold Limit Values/Time Weighted Averages (TLV/TWA), or value for a similar toxicological end point from toxicological data in the Registry of Toxic Effects for Chemical Substances (RTEC).

b. ERPG 2 (irreversible health effects), or 0.1 of Immediately Dangerous to Life and Health (IDLH), or value for a similar toxicological end point from toxicological data in RTEC.

c. ERPG 3 (death), IDLH, or value for a similar toxicological end point from toxicological data in RTEC.

d. Bold italic type indicates that the toxicological limit was exceeded at one or more exposure points.

Table 5.15-9. Estimated injuries, illnesses, and fatalities of workers expected during construction and operation of facilities in each alternative (cumulative totals through 2035).

Alternative	Construction Workers[a]		Operations Workers[a]		Total Workers	
	Injury & illness (persons)	Fatalities (persons)	Injury & illness (persons)	Fatalities (persons)	Injury & illness (persons)	Fatalities (persons)
No Action[b]	0	0	231	0	231	0
Decentralization						
Suboption W	54	0	83	0	137	0
Suboption X	49	0	84	0	133	0
Suboption Y[c]	79	0	69	0	148	0
Suboption Z[c]	48	0	69	0	117	0
Suboption P[c]	183	0	84	0	267	0
Suboption Q[c]	223	0	139	0	362	1
1992/3 Planning Basis	same as Decentralization					
Regionalization						
Suboption AX	38	0	82	0	120	0
Suboption AY[c]	74	0	69	0	143	0
Suboption AZ[c]	37	0	69	0	106	0
Suboption B1[d]	99	0	109	0	208	0
Suboption B2[d]	211	0	136	0	347	1
Suboptions C	same as Centralization offsite					
Centralization Onsite[d]	285	0	205	0	490	1
Centralization Offsite	154	0	84	0	238	0

a. Facility construction and operation estimates are based on DOE and DOE contractor accident rates (See Volume 2, Part B, Table F-4-7 of this EIS).
b. Worker year estimates from Bergsman (1995).
c. Dry storage suboptions (Y or Z) would be paired with either of two processing options (P or Q).
d. These estimates represent incremental increases for fuel imported from offsite locations only; estimates for storage (and stabilization where required) of onsite fuel woule be the same as in the Decentralization Alternative.

5.16.1.2 Air Quality. Air quality limits (WAC 173-470-030,-100) at the Hanford Site boundary are not expected to be approached as a result of implementing the No Action Alternative or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory or from decommissioning of unused facilities or site restoration activities.

5.16.1.3 Waste Management. Under the No Action Alternative, there would be a continuing generation of about 100 cubic meters of low-level wastes per year from incidental activities and about 530 cubic meters during containerization of SNF and sludge in the 100-K Area basins. All presently anticipated activities on the Hanford Site would result in approximately 20,000 cubic meters of low-level waste per year. Thus, at a maximum, the total quantity of low-level waste from SNF activities would account for about 5 percent of the annual quantity of low-level waste generated at the Hanford Site.

5.16.1.4 Socioeconomics. Under the No Action Alternative, the SNF workforce would remain the same, about 60 workers. The Hanford Site workforce is expected to drop from about 18,700 in 1995 to 14,700 in 1997 and to remain approximately at 14,700 through 2004. The regional workforce is expected to range from 81,000, to 86,000 in that same period.

5.16.1.5 Occupational and Public Health. The cumulative population dose since plant startup was estimated to be about 100,000 person-rem (estimated to one significant figure; Section 4.12.2.4.2). The number of inferred fatal cancers since plant startup would amount to about 50 (essentially all of which would be attributed to dose received in the 1945-52 time frame). In the 50 years since plant startup, the population of interest (assuming a constant population of 380,000 and an individual dose of about 0.3 rem/year) would have received about 5,000,000 person-rem from naturally occurring radiation sources (natural background) which would relate to about 2,500 latent cancer fatalities. In the same 50 years about 27,000 cancer fatalities from all causes would have been expected in that population.

If the Hanford sitewide contribution to public dose from all exposure pathways is considered (0.8 person-rem per year from DOE facilities and 0.7 person-rem per year from Washington Public Power Supply System reactor operation for 40 years), it is estimated that the cumulative collective dose would be approximately 60 person-rem. No latent fatal cancers would be expected from such a dose. Over 40 years of interim storage of SNF, the population of interest would have received 4,000,000 person-rem from natural background radiation. That dose would relate to 2,000 latent cancer fatalities. In the same 40 years, about 21,000 cancer fatalities from all causes would be expected among the population in the region of interest (380,000 population).

Air quality limits [(40 CFR 61 Subpart H), 10 millirem per year at the Hanford Site boundary] are not expected to be approached as a result of implementing the No Action Alternative or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory or from decommissioning of unused facilities or site restoration activities.

Cumulative spent fuel worker dose from plant startup to date was estimated at about 2,000 person-rem (Section 4.12.1.2), from which one fatal cancer might be inferred. In the near term the annual increments to cumulative worker dose would be expected to be about 24 person-rem. No latent fatal cancers would be expected from 40 years of the No Action Alternative (960 person-rem).

The cumulative worker dose since start up of activities at the Hanford Site is about 90,000 person-rem, to which would be added about 210 person-rem/yr for a total cumulative worker dose of about 100,000 person-rem through the next 40 years. Thus for 90 years of Hanford operations, about 50 latent cancer fatalities (LCFs) might be inferred (4 LCFs inferred from 1995 onward). In those 90 years about 4,500 LCFs would be inferred from natural background radiation and 48,000 LCFs from all causes would be expected.

Although the worker dose assocated with all future site restoration activities is expected to be small in comparison with cumulative worker dose to date, it is too speculative to quantify at this time.

5.16.2 Decentralization Alternative

Cumulative impacts associated with implementation of the Decentralization Alternative are described in the following subsections.

5.16.2.1 Land Use. The Hanford Site consists of about 1450 square kilometers (360,000 acres), of which about 87 square kilometers (22,000 acres) have been disturbed. Implementation of the Decentralization Alternative would disturb an additional area of up to

0.6 square kilometers (160 acres) for a total of about 88 square kilometers (22,000 acres). The amount of land actually occupied by new facilities would range from about 4 ha (11 acres) to about 7 hectares (18 acres). Construction of the Environmental Restoration Disposal Facility will require disturbance of approximately 4.1 square kilometers (1.020 acres) of land. However, restoration of existing disturbed sites will compensate for this loss.

5.16.2.2 Air Quality. Air quality limits (WAC 173-470-030,-100) at the Hanford Site boundary are not expected to be approached as a result of implementing any of the options in the Decentralization Alternative or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory or from decommissioning of unused facilities or restoration activities.

5.16.2.3 Waste Management. In the near term under the Decentralization Alternative, there would be about 530 cubic meters of low-level waste generated during 2 years of repackaging and containerization of SNF and sludge in the 100-K Basins. Thereafter low-level waste generation would range from 41 to 420 cubic meters per year for about 4 years depending on suboption selected. All presently anticipated activities on the Hanford Site would result in approximately 20,000 cubic meters of low-level waste per year. Thus, at a maximum, the total low-level waste from SNF activities would account for about 8 percent of the annual quantity of low-level waste generated at the Hanford Site.

High-level waste that might be generated in the Decentralization Alternative would not add significantly to the more than 250,000 cubic meters of waste at Hanford currently handled as high-level waste.

5.16.2.4 Socioeconomics. Under the Decentralization Alternative, the SNF workforce would increase from 80 to about 740. The Hanford Site workforce is expected to drop from 18,700 in 1995 to 14,700 in 1997 and remain at approximately 14,700 through 2004. The regional workforce is expected to range from 81,000, to 86,000 in that same period. The maximum change with respect to the regional workforce would be an increase of about 0.9 percent.

5.16.2.5 Occupational and Public Health. The cumulative population dose since plant startup was estimated to be about 100,000 person-rem (estimated to one significant figure; Section 4.12.2.4.2). The number of inferred fatal cancers since plant startup would amount to about 50 (essentially all of which would be attributed to dose received in the 1945-52 time frame). In the 50 years since plant startup, the population of interest (assuming a constant population of 380,000 and an individual dose of about 0.3 rem/year) would have received about 5,000,000 person-rem from naturally occurring radiation sources (natural background), which would relate to 2,500 latent cancer fatalities. In the same 50 years about 27,000 cancer fatalities from all causes would have been expected in the region of interest.

If the Hanford sitewide contribution to public dose from all exposure pathways is considered (0.8 person-rem per year from DOE facilities and 0.7 person-rem per year from Washington Public Power Supply System reactor operation for 40 years), it is estimated that the cumulative collective dose would be approximately 60 person-rem. Additional collective population dose from implementation of the Decentralization Alternative would range from 1 to 4 person-rem over 40 years (dose from 4 years of processing would dominate). Thus, in total, the collective population dose from man-made sources would remain approximately 60 person-rem. No latent fatal cancers would be expected from such a dose. Over 40 years of interim storage of SNF, the population of interest would have received 4,000,000 person-rem from naturally occurring radiation sources (natural background). That dose would relate to 2,000 latent cancer fatalities. In the same 40 years, about 21,000 cancer fatalities from all causes would be expected among the population in the region of interest (380,000 population).

Air quality limits [(40 CFR 61 Subpart H), 10 millirem per year at the Hanford Site boundary] are not expected to be approached as a result of implementing the Decentralization Alternative or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory or decommissioning of unused facilities, or site restoration activities.

Cumulative spent fuel worker dose from plant startup to date was estimated at about 2,000 person-rem (Section 4.12.1.2), from which one latent fatal cancer might be inferred. Collective worker dose from SNF activities would amount to about 80 person-rem for maintenance and operations, 18 person-rem for loading storage facilities, and 180 to 320 person-rem depending on processing option selected. Thus, the total collective 40-year worker dose from SNF activities would be from about 300 to 420 person-rem. Within the accuracy of the estimates, cumulative worker dose in the Decentralization Alternative would not add significantly to the cumulative Hanford Site worker dose over 90 years as described for the No Action Alternative.

5.16.3 1992/1993 Planning Basis Alternative

Because of the similarity of activities, cumulative impacts of the 1992/1993 Planning Basis Alternative would be essentially the same as those described for the Decentralization Alternative.

5.16.4 Regionalization Alternative (Options A, B1, B2, and C)

Cumulative impacts for implementation of the four Regionalization Subalternatives are described in the following subsections.

5.16.4.1 Regionalization Option A . Cumulative impacts associated with implementation of the Regionalization Option A where Hanford's defense SNF is stored at the Hanford Site and other SNF is shipped offsite for storage are described in the following subsections.

5.16.4.1.1 Land Use. The Hanford Site consists of about 1450 square kilometers (360,000 acres) of which about 87 square kilometers (22,000 acres) have been disturbed. Implementation of Regionalization Option A would disturb an additional area of up to 0.6 square kilometers (160 acres), for a total of about 88 square kilometers (22,000 acres). The amount of land actually occupied by new facilities would range from about 2 hectares (6 acres) to about 7 hectares (18 acres). Construction of the Environmental Restoration Disposal Facility will require disturbance of approximately 4.1 square kilometers (1.020 acres) of land. However, restoration of existing disturbed sites will compensate for this loss.

5.16.4.1.2 Air Quality. Air quality limits (WAC 173-470-030,-100) at the Hanford Site boundary are not expected to be approached as a result of implementing any of the options in the Regionalization A Alternative or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory or from decommissioning of unused facilities or restoration activities.

5.16.4.1.3 Waste Management. In the near term under Regionalization Option A, there would be about 530 cubic meters of low-level waste generated during containerization of SNF and sludge in the 100-K basins. Thereafter, low-level waste generation would range from 61 to 420 cubic meters per year for about 4 years depending on option selected.. All presently anticipated activities on the Hanford Site would result in approximately

20,000 cubic meters of low-level waste per year. Thus, at a maximum, the total low-level waste from SNF activities would account for about 8 percent of the annual Hanford generation of low-level waste.

High-level waste that might be generated in Regionalization A would not add significantly to the more than 250,000 cubic meters of waste at Hanford currently handled as high-level waste.

5.16.4.1.4 Socioeconomics. Under Regionalization Option A, the SNF workforce would increase by 60 to about 470. The Hanford Site workforce is expected to drop from about 18,700 in 1995 to about 14,700 in 1997 and to remain at approximately 14,700 through 2004. The regional workforce is expected to range from 81,000, to 86,000 in that same period. The maximum change with respect to the regional workforce would be an increase of about 0.6 percent.

5.16.4.1.5 Occupational and Public Health. The cumulative population dose since plant startup was estimated to be about 100,000 person-rem (estimated to one significant figure; Section 4.12.2.4.2). The number of inferred fatal cancers since plant startup would amount to about 50 (essentially all of which would be attributed to exposures in the 1945-52 time frame). In the 50 years since plant startup the population of interest (assuming a constant population of 380,000 and an individual dose of about 0.3 rem/year) would have received about 5,000,000 person-rem from naturally occurring radiation sources (natural background), which would relate to 2,500 latent cancer fatalities. In the same 50 years about 27,000 cancer fatalities from all causes would have been expected in the region of interest.

If the Hanford sitewide contribution to public dose from all exposure pathways is considered (0.8 person-rem per year from DOE facilities and 0.7 person-rem per year from Washington Public Power Supply System reactor operation for 40 years), it is estimated that the cumulative collective dose would be approximately 60 person-rem. Additional collective population dose from implementation of Regionalization Option A would range from 1 to 4 person-rem over 40 years (dose from 4 years of processing would dominate). Thus, in total, the collective population dose from man-made sources would be about 60 person-rem. No latent fatal cancers would be expected from such a dose. Over 40 years of interim storage of SNF, the population of interest would have received 4,000,000 person-rem from naturally occurring radiation sources (natural background). That dose would relate to 2,000 latent cancer fatalities.

In the same 40 years, about 21,000 cancer fatalities from all causes would be expected among the population in the region of interest (380,000 population).

Air quality limits ([40 CFR 61 Subpart H], 10 millirem per year at the Site boundary) are not expected to be approached as a result of implementing the Regionalization Alternative or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory, or decommissioning of unused facilities, or site restoration activities.

Cumulative spent fuel worker dose from plant startup to date was estimated at about 2,000 person-rem (Section 4.12.1.2), from which one latent fatal cancer might be inferred. Collective worker dose from SNF activities would amount to about 80 person-rem for maintenance and operations, 18 person-rem for loading storage facilities, and 180 to 320 person-rem depending on processing option selected. Thus the total collective 40-year worker dose would be from about 300 to 420 person-rem. Within the accuracy of the estimates, cumulative worker dose in Regionalization A would not add significantly to the cumulative Hanford Site work dose over 90 years as described for the No Action Alternative.

5.16.4.2 Regionalization Option B1. Cumulative impacts associated with the implementation of Regionalization Option B1, where all SNF west of the Mississippi River, except for Naval SNF, is transported to Hanford are described in the following subsections.

5.16.4.2.1 Land Use. The Hanford Site consists of about 1450 square kilometers (360,000 acres), of which about 87 square kilometers (22,000 acres) have been disturbed. Implementation of Regionalization Option B1 would disturb an additional area of upto 0.6 square kilometers (160 acres), for a total of about 88 square kilometers (22,000 acres). The amount of land actually occupied by new facilities would range from about 15 hectares (36 acres) to about 28 hectares (68 acres). Construction of the Environmental Restoration Disposal Facility will require disturbance of approximately 4.1 square kilometers (1.020 acres) of land. However, restoration of existing disturbed sites will compensate for this loss.

5.16.4.2.2 Air Quality. Air quality limits (WAC 173-470-030,-100) at the Hanford Site boundary are not expected to be approached as a result of implementing any of the options in Regionalization Option B1 or from reasonably foreseeable additions to the

Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory or from decommissioning of unused facilities or restoration activities.

5.16.4.2.3 Waste Management. In the near term under Regionalization Option B1, there would be about 530 cubic meters of low-level waste generated during repackaging and containerization of SNF and sludge in 100-K Basins. Thereafter low-level waste generation would range from 61 to 420 cubic meters per year for about 4 years depending on the suboption selected. All presently anticipated processing activities on the Hanford Site would result in approximately 20,000 cubic meters of low-level waste per year. Thus, the total quantity of low-level waste from SNF activities would account for about 8 percent of the annual quantity of low-level waste generated at the Hanford Site.

High-level waste that might be generated in Regionalization B1 would not add significantly to the more than 250,000 cubic meters of waste at Hanford currently handled as high-level waste.

5.16.4.2.4 Socioeconomics. Under Regionalization Option B1, the SNF workforce would increase by about 170 to about 800. The Hanford Site workforce is expected to drop from 18,700 in 1995 to 14,700 in 1997 and remain around 14,700 through 2004. The regional workforce is expected to range from 81,000, to 86,000 in that same period. The maximum change with respect to the regional workforce would be an increase of about 1 percent.

5.16.4.2.5 Occupational and Public Health. The cumulative population dose since plant startup was estimated to be about 100,000 person-rem (estimated to one significant figure; Section 4.12.2.4.2). The number of inferred fatal cancers since plant startup would amount to about 50 (essentially all of which would be attributed to exposures in the 1945-52 time frame). In the 50 years since plant startup, the population of interest (assuming a constant population of 380,000) would have received about 5,000,000 person-rem from naturally occurring radiation sources (natural background), which would relate to 2,500 latent cancer fatalities. In the same time, about 27,000 cancer fatalities from all causes would have been expected in the region of interest.

If the Hanford sitewide contribution to public dose from all exposure pathways is considered (0.8 person-rem per year from DOE facilities and 0.7 person-rem per year from

Washington Public Power Supply System reactor operation for 40 years), it is estimated that the cumulative collective dose would be approximately 60 person-rem. Additional collective population dose from implementation of Regionalization Option B1 would range from 1 to 4 person-rem over 40 years (dose from 4 years of processing would dominate). Thus, in total, the collective population dose from man-made sources would remain approximately 60 person-rem. No latent fatal cancers would be expected from such a dose. Over 40 years of interim storage of SNF, the population of interest would have received 4,000,000 person-rem from naturally occurring radiation sources (natural background). That dose would relate to 2,000 latent cancer fatalities. In the same 40 years, about 21,000 cancer fatalities from all causes would be expected among the population in the region of interest (380,000 population).

Air quality limits [(40 CFR 61 Subpart H), 10 millirem per year at the Hanford Site boundary] are not expected to be approached as a result of implementing Regionalization Option B1 or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory or from decommissioning of unused facilities or site restoration activities.

Cumulative spent fuel worker dose from plant startup to date was estimated at about 2,000 person-rem (Section 4.12.1.2), from which one latent fatal cancer might be inferred. Collective worker dose from SNF activities would amount to about 80 person-rem for maintenance and operations, 18 person-rem for loading storage facilities, and 180 to 320 person-rem depending on processing option selected. Thus the total collective 40-year worker dose would be from about 300 to 420 person-rem. Within the accuracy of the estimates, cumulative worker dose in Regionalization B1 would not add significantly to the cumulative Hanford Site worker dose over 90 years as described for the No Action Alternative.

5.16.4.3 Regionalization Option B2. Cumulative impacts associated with the implementation of Regionalization Option B2, where all SNF west of the Mississippi River and Naval SNF, are transported to Hanford are described in the following subsections.

5.16.4.3.1 Land Use. The Hanford Site consists of about 1450 square kilometers (360,000 acres) of which about 87 square kilometers (22,000 acres) have been disturbed. Implementation of Regionalization Option B2 would disturb an additional area of up to 0.6 square kilometers (160 acres), for a total of about 88 square kilometers (22,000 acres). The amount of land actually occupied by new facilities would range from about 21 hectares

(52 acres) to about 30 hectares (74 acres). Construction of the Environmental Restoration Disposal Facility will require disturbance of approximately 4.1 square kilometers (1.020 acres) of land. However, restoration of existing disturbed sites will compensate for this loss.

5.16.4.3.2 Air Quality. Air quality limits (WAC 173-470-030,-100) at the Hanford Site boundary are not expected to be approached as a result of implementing any of the suboptions in Regionalization Option B1 or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory, or from decommissioning of unused facilities or restoration activities.

5.16.4.3.3 Waste Management. In the near term under Regionalization Option B2, there would be about 530 cubic meters of low-level waste generated during repackaging and containerization of SNF and sludge in the 100-K Basins. Thereafter, low-level waste generation would range from 61 to 420 cubic meters per year. All presently anticipated activities on the Hanford Site would result in approximately 20,000 cubic meters of low-level waste per year. Thus, at a maximum, the total quantity of low-level waste from SNF activities would account for about 4 percent of the annual quantity of low-level waste generated at the Hanford Site.

High-level waste that might be generated in Regionalization B2 would not add significantly to the more than 250,000 cubic meters of waste at Hanford currently handled as high-level waste.

5.16.4.3.4 Socioeconomics. Under Regionalization Option B2, the SNF workforce would increase by about 170 to about 800. The Hanford Site workforce is expected to drop from 18,700 in 1995 to 14,700 in 1997 and remain around 14,700 through 2004. The regional workforce is expected to range from 81,000, to 86,000 in that same period. The maximum change with respect to the regional workforce would be an increase of about 1 percent.

5.16.4.3.5 Occupational and Public Health. The cumulative population dose since plant startup was estimated to be about 100,000 person-rem (estimated to one significant figure; Section 4.12.2.4.2). The number of inferred fatal cancers since plant startup would amount to about 100 (essentially all of which would be attributed to exposures in the 1945-52 time frame). In the 50 years since plant startup, the population of interest (assuming a constant population of 380,000) would have received about 5,000,000 person-rem from naturally occurring radiation sources (natural background) which would relate to 2,500 latent cancer fatalities. In the same time about 27,000 cancer fatalities from all causes would have been expected in the region of interest.

If the Hanford Site contribution from all exposure pathways to public dose is added (0.8 person-rem per year from DOE facilities and 0.7 person-rem per year from Washington Public Power Supply System reactor operation for 40 years), it is estimated that the cumulative collective dose would be approximately 60 person-rem. Additional collective population dose from implementation of Regionalization Option B2 would range from 1 to 4 person-rem over 40 years (dose from 4 years of processing would dominate). Thus, in total, the collective population dose from man-made sources would remain approximately 60 person-rem. No latent fatal cancers would be expected from such a dose. Over 40 years of interim storage of SNF, the population of interest would have received 4,000,000 person-rem from naturally occurring radiation sources (natural background). That dose would relate to 2,000 latent cancer fatalities. In the same 40 years, about 21,000 cancer fatalities from all causes would be expected among the population in the region of interest (380,000 population).

Air quality limits [(40 CFR 61 Subpart H), 10 millirem per year at the Site boundary] are not expected to be approached as a result of implementing Regionalization Option B2 or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory, or decommissioning of unused facilities or site restoration activities.

Cumulative spent fuel worker dose from plant startup to date was estimated at about 2,000 person-rem (Section 4.12.1.2), from which one latent fatal cancer might be inferred. Collective worker dose from SNF activities would amount to about 80 person-rem for maintenance and operations, 18 person-rem for loading storage facilities, and 180 to 320 person-rem depending on the processing suboption selected. Thus the total collective 40-year worker dose would be from about 300 to 420 person-rem. Within the accuracy of the estimates, cumulative worker dose in Regionalization B2 would not add significantly to the cumulative Hanford Site worker dose over 90 years as described for the No Action Alternative.

5.16.4.4 Regionalization C Option. Cumulative impacts in this option, where all Hanford SNF is sent to INEL or NTS, would be essentially the same as those described for the Centralization Alternative, minimum option.

5.16.5 Centralization Alternative

Cumulative impacts associated with implementation of one or the other of two options under the Centralization Alternative are described in the following subsections.

5.16.5.1 Centralization Alternative Maximum Option. Cumulative impacts associated with implementation of the Centralization Alternative maximum option, where all SNF is sent to the Hanford Site, are described in the following subsections.

5.16.5.1.1 Land Use. The Hanford Site consists of about 1450 square kilometers (360,000 acres), of which about 87 square kilometers (22,000 acres) have been disturbed. Implementation of the Centralization Alternative maximum option would disturb up to an additional area of about 0.6 square kilometers (160 acres) for a total of about 88 square kilometers (22,000 acres). The amount of land actually occupied by new facilities would range from about 35 hectares (86 acres) to about 38 hectares (93 acres). Construction of the Environmental Restoration Disposal Facility will require disturbance of approximately 4.1 square kilometers (1.020 acres) of land. However, restoration of existing disturbed sites will compensate for this loss.

5.16.5.1.2 Air Quality. Air quality limits (WAC 173-470-030,-100) at the Hanford Site boundary are not expected to be approached as a result of implementing any of the suboptions in the Centralization Alternative maximum option or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory, or from decommissioning unused facilities or restoration activities.

5.16.5.1.3 Waste Management. In the near term under the Centralization Alternative maximum option, there would be about 532 cubic meters of low-level waste generated during repackaging and containerization of SNF and sludge in the 100-K Basins. Thereafter, low-level waste generation would amount to about 140 cubic meters per year. All presently anticipated activities on the Hanford Site would result in approximately 20,000 cubic meters of low-level waste per year. Thus, at a maximum, SNF activities would account for about 1 percent of the total.

High-level waste that might be generated in the Centralization maximum option would not add significantly to the more than 250,000 cubic meters of waste at Hanford currently handled as high-level waste.

5.16.5.1.4 Socioeconomics. Under the Centralization Alternative maximum option, the SNF workforce would increase by about 290 to about 900. The Hanford Site workforce is expected to drop from 18,700 in 1995 to 14,700 in 1997 and remain around 14,700 through 2004. The regional workforce is expected to range from 81,000, to 86,000 in that same period. The maximum change with respect to the regional workforce would be an increase of about 1 percent.

5.16.5.1.5 Occupational and Public Health. The cumulative population dose since plant startup was estimated to be about 100,000 person-rem (estimated to one significant figure; Section 4.12.2.4.2). The number of inferred fatal cancers since plant startup would amount to about 50 (essentially all of which would be attributed to exposures in the 1945-52 time frame). In the 50 years since plant startup, the population of interest (assuming a constant population of 380,000) would have received 5,000,000 person-rem from naturally occurring radiation sources (natural background), which would relate to 2,500 latent cancer fatalities. In the same time about 27,000 cancer fatalities from all causes would have been expected in the region of interest .

If the Hanford sitewide contribution to public dose from all exposure pathways is considered (0.8 person-rem per year from DOE facilities and 0.7 person-rem per year from Washington Public Power Supply System reactor operation for 40 years), it is estimated that the cumulative collective dose would be approximately 60 person-rem. Additional collective population dose from implementation of the Centralization Alternative maximum option would range from 1 to 4 person-rem over 40 years (dose from 4 years of processing would dominate). Thus, in total, the collective population dose from man-made sources would remain approximately 60 person-rem. No latent fatal cancers would be expected from such a dose. Over 40 years of interim storage of SNF, the population of interest would have received 4,000,000 person-rem from naturally occurring radiation sources (natural background). That dose would relate to 2,000 latent cancer fatalities. In the same 40 years, about 21,000 cancer fatalities from all causes would be expected among the population in the region of interest (380,000 population).

Air quality limits [(40 CFR 61 Subpart H), 10 millirem per year at the Hanford Site boundary] are not expected to be approached as a result of implementing the Centralization Alternative maximum option or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory, or decommissioning of unused facilities or site restoration activities.

Cumulative spent fuel worker dose from plant startup to date was estimated at about 2,000 person-rem (Section 4.12.1.2), from which one latent fatal cancer might be inferred. Collective worker dose from SNF activities in the Centralization Alternative maximum option would amount to about 80 person-rem for maintenance and operations, 18 person-rem for loading storage facilities, and 180 to 320 person-rem depending on processing suboption selected.

Within the accuracy of the estimates, cumulative worker dose in the Centralization maximum option would not add significantly to the cumulative Hanford Site worker dose over 90 years as described for the No Action Alternative.

5.16.5.2 Centralization Alternative Minimum Option. Cumulative impacts associated with implementation of the Centralization Alternative minimum option, where all SNF on the Hanford Site is shipped offsite for storage, are described in the following subsections.

5.16.5.2.1 Land Use. The Hanford Site consists of about 1450 square kilometers (360,000 acres) of which about 87 square kilometers (22,000 acres) have been disturbed. Implementation of the Centralization Alternative minimum option would disturb up to an additional area of about 0.6 square kilometers (160 acres) for a total of about 88 square kilometers (22,000 acres). The amount of land actually occupied by new facilities would range from about 2 hectares (6 acres) to about 15 hectares (12 acres). Construction of the Environmental Restoration Disposal Facility will require disturbance of approximately 4.1 square kilometers (1.020 acres) of land. However, restoration of existing disturbed sites will compensate for this loss.

5.16.5.2.2 Air Quality. Air quality limits (WAC 173-470-030,-100) at the Hanford Site boundary are not expected to be approached as a result of implementing the any of the suboptions in the Centralization Alternative minimum option or from reasonably foreseeable additions to the Hanford Site, e.g., construction and operation of a Laser Interferometer Gravitational-Wave Observatory, or from decommissioning unused facilities or restoration activities.

5.16.5.2.3 Waste Management. In the near term under the Centralization Alternative minimum option, there would be about 532 cubic meters of low-level waste generated during repackaging and containerization of SNF and sludge in the 100-K Basins. Thereafter, low-level waste generation would range from 110 to 490 cubic meters per year. All presently anticipated activities on the Hanford Site would result in approximately 21,000 cubic meters of solid waste per year. Thus, at a maximum, SNF activities would account for about 2 percent of the annual generation of low-level waste at the Hanford Site.

High-level waste that might be generated in the Centralization mininim option would not add significantly to the more than 250,000 cubic meters of waste at Hanford currently handled as high-level waste.

5.16.5.2.4 Socioeconomics. Under the Centralization Alternative minimum option, the SNF workforce would increase by about 390 to about 590. The Hanford Site workforce is expected to remain at about 18,000 from 1995 through 2004. The regional workforce is expected to range from 81,000, to 86,000 in that same period. The maximum change with respect to the regional workforce would be an increase of about 0.7 percent.

5.16.5.2.5 Occupational and Public Health. The cumulative population dose since plant startup was estimated to be about 200,000 person-rem (estimated to one significant figure; Section 4.12.2.4.2). The number of inferred fatal cancers since plant startup would amount to about 50 (essentially all of which would be attributed to exposures in the 1945-52 time frame). In the 50 years since plant startup, the population of interest (assuming a constant population of 380,000) would have received 5,000,000 person-rem from naturally occurring radiation sources (natural background), which would relate to 2,500 latent cancer fatalities. In the same time about 24,000 cancer fatalities from all causes would have been expected in the region of interest.

Cumulative spent fuel worker dose from plant startup to date was estimated at about 2,000 person-rem (Section 4.12.1.2), from which one latent fatal cancer might be inferred. Collective worker dose from SNF activities in the Centralization Alternative minimum option would amount to about 80 person-rem for maintenance and operations, 18 person-rem for loading storage facilities, and 180 to 320 person-rem depending on processing suboption selected. Thus the total collective 40-year worker dose would be from about 300 to 420 person-rem.

Within the accuracy of the estimates, cumulative worker dose in the Centralization minimum option would not add significantly to the cumulative Hanford Site worker dose over 90 years as described for the No Action Alternative.

5.17 Adverse Environmental Impacts that Cannot be Avoided

Unavoidable adverse impacts that might arise as a result of implementing the alternatives for interim storage of SNF at the Hanford Site are discussed in the following subsections.

5.17.1 No Action Alternative

Adverse impacts associated with the No Action Alternative would derive from the expense and radiation exposure associated with maintaining facilities that are near or at the end of their design life and the possible future degradation of fuel and facilities, thus increasing the potential for releases of materials to the environment.

5.17.2 Decentralization Alternative

Adverse impacts associated with the Decentralization Alternative would derive principally from construction activities needed for new facilities. There would be displacement of some animals from the construction site and the destruction of plant life within the site up to 9 hectares (24 acres). Criteria pollutants, radionuclides, and hazardous chemicals would also be released in up to permitted quantities during processing preparations. Traffic congestion and noise are expected to increase by a few percent during the construction of major facilities. Competition for adequate housing would increase in the already tight market, and capacities at some of the local school would be moderately strained with approximately 0.5 to 1.5 percent additional students, depending on which processing and/or storage option were chosen.

5.17.3 1992/1993 Planning Basis Alternative

Adverse impacts associated with the 1992/1993 Planning Basis Alternative would be essentially the same as those for the Decentralization Alternative. If transport of any amount of SNF were considered an adverse impact, that impact would occur in this alternative if the small amount of TRIGA fuel at Hanford were transported to INEL.

5.17.4 Regionalization Alternative

Unavoidable adverse environmental impacts for the Regionalization Alternative range from those of the Centralization (Minimum) Alternative for Regionalization C where all Hanford SNF is shipped offsite to essentially those of the Centralization (Maximum) Alternative for Regionalization B2 where all SNF west of the Mississippi River including Naval SNF is shipped to Hanford.

5.17.5 Centralization Alternative

In the option where Hanford receives all DOE SNF, adverse impacts would be somewhat larger than those associated with implementing the Decentralization Alternative because about 25 weight percent more fuel than already exists on the Hanford Site would need to be stored; however, higher heat loads on that fuel might nearly triple the capacity needed for storage. Transport of that 25 weight percent of SNF to the Hanford Site also likely would be viewed as an adverse impact.

In the option where Hanford ships all of its fuel to another site, adverse impacts would be associated with construction and operation of a fuel packaging facility. The impacts, however, would be expected to be substantially less than those noted for the Decentralization Alternative. Transporting a relatively large amount of SNF offsite to another DOE facility also likely would be considered an adverse impact.

5.18 Relationship Between Short-Term Uses of the Environment and the Maintenance and Enhancement of Long-Term Productivity

SNF storage is contemplated for up to 40 years pending decisions on ultimate disposition. SNF is essentially uranium-238 with varying amounts of uranium-235 and small amounts of plutonium contaminated by small masses of fission products (but high activity). Because of this composition, a decision could be made at the end of the planned storage period to either continue storage until the energy resource value of the SNF warrants processing for power-reactor fuel or to determine that the fuel will never have any resource value and will be disposed of. If the decision is to continue to store the SNF, that option could be seen as the best use of land at the Hanford Site in terms of long-term productivity. This conclusion would apply to all of the alternatives except for the Regionalization C Alternative and the Centralization Alternative with storage at other than Hanford.

If the decision is to dispose of the SNF or if the non-Hanford centralization option for storage is selected, the land on the Hanford Site would become available for other uses. Because of the potential for, or perception of, contamination, use of the land for agriculture might not be appropriate. Moreover, the land occupied (or that would be occupied) by SNF facilities was of marginal utility for farming before it was obtained for the Hanford Site, and it remains so. However, other uses, such as for wildlife refuges, might be appropriate long-term uses of land vacated by SNF facilities after decommissioning is completed.

5.19 Irreversible and Irretrievable Commitment of Resources

This section addresses the irretrievable commitment of resources that would likely be used to implement the proposed project or its alternatives. An irretrievable resource is a natural or physical resource that is irreplaceably lost and cannot be replenished.

Implementation of the proposed project would result in the irretrievable use of fossil fuels in construction activities and in the transport of raw materials to the project site. In addition, there would be an irretrievable use of electricity and fossil fuel in the SNF operations. Briefly summarized below are discussions of irretrievable and irreversible resource impacts for each alternative.

5.19.1 No Action Alternative

The irreversible and irretrievable commitment of resources for the No Action Alternative would include an additional increment of energy, materials, and manpower to maintain safe and secure facilities. A new SNF facility would not be built, and Hanford SNF would continue to be managed in the current mode.

If the No Action Alternative were implemented, the following facilities would likely be used at the Hanford Site to maintain continued safe and secure storage of SNF: the 105-KE and KW Basins, FFTF, T-Plant, and the 308, 324, 325, and 327 buildings. Excluding energy and materials expended during construction of minor facilities to maintain safety and security, the operational staff is estimated at 215 personnel, and electrical power consumption is estimated to be 12,000 megawatt hours per year. This alternative represents less than a 2 percent increase in existing personnel at the Hanford Site and a negligible increase in the total amount of electrical energy currently used at the Hanford Site.

5.19.2 Decentralization Alternative

The irreversible and irretrievable commitment of resources for the Decentralization Alternative would include an additional increment of energy, materials, and personnel. Existing Hanford Site SNF would be safely stored for a 40-year period, with some limited SNF shipments. To accommodate this mission, existing facilities would require upgrading and new storage systems would need to be constructed. Various options have been proposed on which facilities to build and how to upgrade existing ones, but it has not been determined exactly which kind of facilities would need to be built. A representative set of values is presented in Table 5.19-1, which roughly indicates the material, personnel, and energy commitments. Depending on the option chosen, the alternative could require less than a 1.5 percent increase or up to a 33 percent increase (but only for 4 years) in the total amount of electrical energy currently used at the Hanford Site.

In addition to energy increases, additional water resources would be required for this alternative, but are not expected to be an excessive amount, compared to the more than 15 million cubic meters (4 billion gallons) of water used each year on the Hanford Site for all processes.

Table 5.19-1. Irretrievable commitment of materials in the Decentralization Alternative suboptions.

Item	Suboption					
	W	X	Y	Z	P	Q
Concrete, thousand cubic meters/(cubic yards)	13 (17)	15 (20)	17 (23)	24 (32)	22 (29)	29 (38)
Lumber, thousand cubic meters (board feet)	1.2 (500)	1.4 (570)	1.6 (650)	2.2 (930)	2.0 (850)	2.6 (1100)
Electricity						
Construction (MW--hrs)	2500	2900	3500	4800	4370	5700
Operations (MW-hrs/yr)	1600	1600	100	100	40,000	127,000
Diesel fuel, cubic meters (thousand gallons)	500 (130)	570 (150)	660 (175)	900 (240)	830 (220)	1100 (290)
Gasoline, cubic meters (thousand gallons)	500 (130)	570 (150)	660 (175)	900 (240)	830 (220)	1100 (290)

a. Assumes operation of the process facility (28,000 or 115,000 MW-Hrs/yr) concurrently with those facilities where SNF is currently stored (12,000 MW-Hrs/yr, as in the No Action Alternative) for an interim period less than 4 years.

5.19.3 1992/1993 Planning Basis Alternative

The irreversible and irretrievable commitment of resources for the 1992/1993 Planning Basis Alternative would be very similar to those for the Decentralization Alternative. The materials, personnel, and energy estimates are assumed to approximate those stated in the Decentralization Alternative.

5.19.4 Regionalization Alternative

The Regionalization Alternative as it applies to the Hanford Site contains the following options:

- Option A - All SNF except defense production SNF would be sent to INEL.
- Option B1 - All SNF west of the Mississippi River except Naval SNF would be sent to Hanford.
- Option B2 - All SNF west of the Mississippi River and Naval SNF would be sent to Hanford.
- Option C - All Hanford SNF would be sent to INEL or NTS.

With the exception of Option C, which for Hanford is equivalent to the Centralization Alternative minimum option, the irretrievable and irreversible commitment of material resources are provided in Tables 5.19-2 through 5.19-4.

5.19.5 Centralization Alternative

The Centralization Alternative has two major options: either all Hanford SNF would be shipped offsite to another DOE facility where all SNF would be centralized (minimum option), or the Hanford Site would become the centralized location for all DOE SNF to be temporarily

Table 5.19-2. Irretrievable commitment of material resources in the Regionalization A suboptions.

Item	Suboption					
	W	X	Y	Z	P	Q
Concrete, thousand cubic meters/(cubic yards)	9 (12)	9 (12)	16 (21)	19 (25)	22 (29)	29 (38)
Lumber, thousand cubic meters (board feet)	0.8 (350)	0.8 (350)	1.4 (600)	1.7 (700)	2.0 (850)	2.6 (1100)
Electricity						
Construction (MW-hrs)	1800	1800	3200	3800	4370	5700
Operations (MW-hrs/yr)	1600	1600	100	100	40,000[a]	127,000[a]
Diesel fuel, cubic meters (thousand gallons)	380 (100)	380 (100)	610 (160)	720 (190)	830 (220)	1100 (290)
Gasoline, cubic meters (thousand gallons)	380 (100)	380 (100)	610 (160)	720 (190)	830 (220)	1100 (290)

a. Assumes operation of the process facility (28,000 or 115,000 MW-Hrs/yr) concurrently with those facilities where SNF is currently stored (12,000 MW-Hrs/yr, as in the No Action Alternative) for an interim period less than 4 years.

Table 5.19-3. Irretrievable commitment of material resources in the Regionalization B1 option. (In addition to those listed for the Decentralization Alternative)

Concrete, thousand cubic meters/(cubic yards)	54 (70)
Lumber, thousand cubic meters (board feet)	5 (2,000)
Electricity, megawatt hours per year	3.000
Diesel fuel, cubic meters (thousand gallons)	1,900 (500)
Gasoline, cubic meters (thousand gallons)	1,900 (500)

Table 5.19-4. Irretrievable commitment of material resources in the Regionalization B2 option. (In addition to those listed for the Decentralization Alternative)

Concrete, thousand cubic meters/(cubic yards)	120 (150)
Lumber, thousand cubic meters (board feet)	10 (4,200)
Electricity, megawatt hours per year	3,000
Diesel fuel, cubic meters (thousand gallons)	4,400 (1,200)
Gasoline, cubic meters (thousand gallons)	4,400 (1,200)

stored (maximum option). The increases in energy, materials, and personnel for both options are shown in Table 5.19-5. If all the SNF were shipped to the Hanford Site, then the impacts would be similar, although somewhat larger, than those of the Regionalization B options. If all the SNF were shipped offsite, then the impacts would be identical to the similar Regionalization B options. If all SNF were shipped offsite, construction and operation of a fuel packaging facility would be necessary before shipments could be made to an offsite facility.

5.20 Potential Mitigation Measures

This section summarizes possible mitigation measures that might be considered to avoid or reduce impacts to the environment as a result of Hanford Site operations in support of SNF management. These measures would be reviewed and revised as appropriate, depending on the specific actions to be taken at a facility, the level of impact, and other pertinent factors.

Table 5.19-5. Irretrievable commitment of materials in the Centralization options.

Item	No Fuel Stored at the Hanford Site	All Offsite Fuel Stored at the Hanford Site
Concrete, thousand cubic meters (cubic yards)	18 (23)	150 (200)
Lumber, thousand cubic meters (board feet)	1.6 (660)	13 (5600)
Electricity, megawatt hours per year	0-20,000	100-127,000
Diesel fuel, cubic meters (thousand gallons)	640 (170)	5700 (1500)
Gasoline, cubic meters (thousand gallons)	640 (170)	5700 (1500)

Possible mitigation measures are generally the same for all alternatives and are summarized by resource category below. No impacts on land use and aesthetic and scenic resources were identified; therefore, mitigation measures would not be necessary.

5.20.1 Pollution Prevention/Waste Minimization

The U.S. Department of Energy is responding to Executive Order 12856 and associated DOE orders and guidelines by reducing the use of toxic chemicals; improving emergency planning, response, and accident notification; and encouraging the development and use of clean technologies and the testing of innovative pollution prevention technologies. Program components include waste minimization, source reduction and recycling, and procurement practices that preferentially procure products made from recycled materials. The pollution prevention program at the Hanford Site is formalized in a Hanford Site Waste Minimization and Pollution Prevention Awareness Program Plan.

The SNF program activities would be conducted in accordance with this plan and implementation of the pollution prevention and waste minimization plans would minimize the generation of waste during SNF management activities.

5.20.2 Socioeconomics

The level of predicted employment for SNF activities at the Hanford Site is not large enough in comparison with present Hanford, local, or regional employment to produce a boom-bust impact on the economy.

5.20.3 Cultural (Archaeological, Historical, and Cultural) Resources

To avoid loss of cultural resources during construction of SNF facilities on the Hanford Site a cultural resources survey of the area of interest would be conducted by PNL Cultural Resources staff. Assuming no such resources were found, construction would proceed. If, however, during construction (earth moving) any cultural resource is discovered, construction activities would be halted and the PNL Cultural resources staff called upon to evaluate and determine the appropriate disposition of the find.

To avoid loss of cultural resources during operation, such as unauthorized artifact collection, workers could be educated through programs and briefing sessions to inform them of applicable laws and regulations for site protection. These educational programs would stress the importance of preserving cultural resources and specifics of the laws and regulations for site protection. The exact location of cultural resources are not identified by the PNL Cultural Resources group; therefore, any such artifact collection would be in an area discovered by the worker(s).

5.20.4 Geology

Soil loss would be controlled during construction using standard dust suppression techniques on disturbed soil and by stockpiling with cover where necessary. Following construction, soil loss would be controlled by revegetation and relandscaping of disturbed areas. Any soil that might become contaminated as a result of SNF management activities could be remediated using methods appropriate to the type and extent of contamination.

5.20.5 Air Resources

To avoid impacts associated with emissions of fugitive dust during construction activities, exposed soils would be treated using standard dust suppression techniques. New facility sources of pollutant emissions to the atmosphere would be designed using best available technology to reduce emissions to as low as reasonably achievable.

5.20.6 Water Resources

The impacts to surface and groundwater sources could be minimized through recycling of water, where feasible, and with clean-up of excess process water before release to ground or surface water.

5.20.7 Ecology

To avoid impacts to endangered, candidate, or state-identified sensitive species, pre-construction surveys would be completed to determine the presence of these species or their habitat. Within six months of ground breaking, DOE would again consult with the U.S. Fish and Wildlife Service to determine current species listings and perform a biological survey of the proposed SNF site. The presently proposed site at Hanford has been surveyed and no currently listed species were found. While not endangered, stands of Big Sagebrush habitat are diminishing generally and Hanford would expect to implement its habitat replacement program to provide areas on at least a 2 to 1 basis to mitigate habitat loss. In addition, areas disturbed would, as appropriate, be seeded with native plant species.

5.20.8 Noise

Generation of construction and operations noise would be reduced, as practicable, by using equipment that complies with EPA noise guidelines (40 CFR Parts 201-211). Construction workers and other personnel working in environments exceeding EPA-recommended guidelines during SNF storage construction or operation would be provided with earmuffs or earplugs approved by the Occupational Safety and Health Administration (29 CFR Part 1910). Because of the remote location of the Hanford SNF activities, there would be no noise impacts with respect to the public for which mitigation would be necessary.

5.20.9 Traffic and Transportation

At sites with increasing traffic concerns, DOE could encourage use of high-occupancy vehicles (such as vans or buses), implementing carpooling and ride-sharing programs, and staggering workhours to reduce peak traffic.

5.20.10 Occupational and Public Health and Safety

Although no radiological impacts on workers or the public were evident from the evaluation of routine SNF activities at Hanford, further improvement in controls to protect both workers and the general public is a continuing activity. The as low as reasonably achievable (ALARA) principle would be used for controlling radiation exposure and exposure to hazardous/toxic substances. Hanford would continue to refine its current emergency planning, emergency preparedness, and emergency response programs in place to protect both workers and the public.

5.20.11 Site Utilities and Support Services

No mitigation measures beyond those identified for ground disturbance activities associated with bringing power and water to the SNF site would appear necessary. In those cases use of standard dust suppression techniques and revegetation of disturbed areas would mitigate ground disturbance impacts.

5.20.12 Accidents

The Hanford Site maintains an emergency response center and has emergency action plans and equipment to respond to accidents and other emergencies. These plans include training of workers, local emergency response agencies (such as fire departments) and the public communication systems and protocols, readiness drills, and mutual aid agreements. The plans would be updated to include consideration of new SNF facilities and activities. Design of new facilities to current seismic and other facility protection standards would reduce the potential for accidents, and implementation of emergency response plans would substantially mitigate the potential for impacts in the event of an accident.

6. LIST OF PREPARERS

Rosanne L. Aaberg, dose calculations. B.S. (Chemical Engineering) University of Washington. Seventeen years of experience in dose calculations, and EIS preparation.

John C. Abbott, affected environment and environmental impacts. B.A. (Geography) Southwest Texas State University, M.S. (Conservation of Natural Resources) University of Texas at San Antonio. Over seventeen years of experience in the preparation of NEPA documents, ecological risk assessment evaluations, regulatory compliance activities, and other program oversight activities.

John M. Alvis, Jr., facility descriptions. B.S. (Nuclear Engineering) and M.S. (Nuclear Engineering) Texas A&M University. Six years of experience in reviewing safety analyses, licensing submittals, and contributing to the development of safety policies and guidance. Assisted in the technical review of licensee documents for NRC.

Larry K. Berg, meteorology. B.S. (Meteorology) Pennsylvania State University. One year of experience in analyzing air quality and air resource parameters.

Frances M. Berting, fuel inventories. B.A. (Physics) Oberlin College, M.A. (Physics) Smith College, Ph.D. (Materials Science) University of Virginia. Characterization of high temperature gas-cooled reactor spent fuel, characterization of N Reactor spent fuel, and experience with non-destructive and destructive examination of irradiated fuel elements. Prepared NRC annual reports on fuel performance at commercial power plants and a report on commercial spent fuel reracking.

Charles A. Brandt, ecological characterization. B.S. (Zoology) Oregon State University, Ph.D. (Zoology) Duke University. Over ten years of experience as a terrestrial ecologist involved in ecological restoration, ecological risk and impact assessment, and conservation biology. Extensive experience in preparation and analysis of NEPA-related documentation.

Mitchel E. Cunningham, spent nuclear fuel management. B.S. (Nuclear Engineering) and M.S. (Nuclear Engineering) Oregon State University. Several years of experience in such projects as the behavior of spent fuel during both inert and air dry storage, investigating in-reactor fission gas release, and the development of integrated computer codes for predicting nuclear fuel rod behavior.

Colbert E. Cushing, deputy project manager, ecological resources. B.S. (Fisheries Management) and M.S. (Limnology) Colorado State University, Ph.D. (Limnology) University of Saskatchewan. Thirty-four years of experience in freshwater ecological research in streams and radioecology, and over twenty years of experience in EIS preparation. Teach university classes in stream ecology and writing journal articles.

Phillip M. Daling, transportation impacts. B.S. (Physical Metallurgy) Washington State University. Related experience includes performing transportation impact calculations for various EIS and environmental assessments and in support of environmental documentation for over ten years.

James F. Donaghue, materials and waste management. B.S. (Civil Engineering) University of Arkansas, J.D. Golden Gate School of Law. Nine years of experience in environmental planning compliance activities. Reviewed EISs and prepared portions of EISs and environmental assessments for Air Force construction projects. Involved in the analysis of alternatives and writing for the DOE Environmental Restoration Programmatic EIS.

Elizabeth A. Flores, materials and waste management. B.S. University of Connecticut, M.A. (Environmental Studies) Yale University. Twelve years of experience in environmental protection and waste management. Assistant Director for Connecticut Department of Environmental Protection for RCRA program.

Stephen Gajewski, regulatory framework and requirements. B.A. (English) and B.A. (Psychology) Gonzaga University. J.D. University of Washington. Over fourteen years of experience in geotechnical operations planning, land management and environmental regulatory compliance, including quality assurance on commercial power reactors, onshore and offshore oil and gas exploration, industrial hygiene program development and training, and environmental strategic planning.

Clifford S. Glantz, non-radiological air quality impacts. B.S. (Physics and Atmospheric Sciences) State University of New York at Albany, M.S. (Atmospheric Sciences) University of Washington. Twelve years of experience in the analysis of non-radiological air quality impacts.

Richard J. Guenther, alternatives and facilities descriptions. B.S. (Engineering Physics), M.S. (Nuclear Engineering), and Ph.D. (Nuclear Engineering) Oregon State University. Over fifteen years of experience testing and evaluating nuclear fuels to determine their characteristics and performance under reactor operating conditions, wet and dry interim storage, and long-term storage in a monitored retrievable or geologic storage environment.

George V. Last, cultural resources and land use. B.S. (Geology) Washington State University. Eighteen years of experience in geological research and cultural resources studies. Extensive experience in preparation and review of NEPA-related documents.

John P. McDonald, water quality and related consequences. A.A.S. (Computer Science) and A.S. (Arts and Science) Columbia Basin College, B.S. (Geology) Eastern Washington University. Four years of experience in conceptual model development of groundwater flow systems, collection of hydraulic head data, and determination of groundwater flow rate and direction, hydraulic testing to determine aquifer properties, testing and maintenance of the waterborne portion of a multiple environmental media computer model, and application of numerical and analytic computer models to environmental problems

Emmett Moore, project manager. B.S. (Chemistry) Washington State University, Ph.D. (Physical Chemistry) University of Minnesota. Twenty years of experience in environmental regulation, participation in and management of the preparation of environmental permits and documentation (NEPA). University professor of physics, chemistry, and environmental sciences.

Iral C. Nelson, deputy project manager, environmental consequences. B.S. (Mathematics) University of Oregon, M.A. (Physics) University of Oregon, diplomate, American Board of Health Physics. Thirty-eight years of experience in various aspects of health physics (radiation protection) and twenty years of experience in conducting NEPA reviews and preparing NEPA documentation.

Ronald C. Phillips, geology and water resources. B.S. (Biology) Wheaton College, M.S. (Botany) Florida State University, Ph.D. (Botany) University of Washington. Wetlands ecologist, including delineation and mitigation of freshwater wetlands. Several years of experience in the preparation and review of categorical exclusion documents, review of environmental assessments, and preparation of biological assessments.

Kathleen Rhoads, air quality and accident analysis. B.S. (Microbiology) and M.S. (Radiological Sciences) University of Washington. Nineteen years of experience in the analysis of risk assessment variables, estimation of radiation does following routine or accidental release of radionuclides to the environment, and evaluation of health effects from energy production.

Chikashi Sato, water quality and related consequences. B.S. (Chemical Engineering) Fukushima National College of Technology, M.S. (Environmental Health Engineering) University of Kansas, Ph.D. (Environmental Engineering) University of Iowa. Thirteen years of experience in university teaching, application of the Multimedia Environmental Pollutant Assessment System (MEPAS), and performance of fate and transport analysis at waste sites.

Dillard B. Shipler, Introduction and review. B.S. (Mathematics and Science) Southern Oregon College, M.S. (Physics) University of Wisconsin-Milwaukee, other studies at University of Oregon, Oregon State University, Reed College, University of Nevada, and University of Washington. More than thirty years of experience in the planning and management of major programs on regulatory compliance, radiological protection, environmental impact assessment, radiological waste management, and environmental safety and health protection.

Donna J. Stucky, socioeconomics. B.A. (Economics) Pacific Lutheran University, M.S. (Agricultural Economics) Purdue University. Two years of experience in the compilation of economic data relating to eastern Washington State.

Betty Tegner, editor. B.A. (English) University of Washington, M.A. (English) California Polytechnical State University. Previous experience in journalism and university teaching. Five years of experience in technical editing.

Gene Whelan, water quality and related consequences. B.S. (Civil Engineering) Pennsylvania State University, M.S. (Mechanics and Hydraulics) University of Iowa, Ph.D. (Civil and Environmental Engineering) Utah State University. Seventeen years of experience in multimedia contaminant environmental exposure assessments.

Mona K. Wright, cultural resources and land use. B.A. (Anthropology) Eastern Oregon State College, M.A. (Anthropology) Washington State University. Fifteen years experience in cultural resource management, Federal regulations including the National Historic Preservation Act, the NEPA, Executive Order 11593, the Archaeological Resources Protection Act of 1979, and the Native American Grave Protection and Repatriation Act, and historic and prehistoric site identification and recording.

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8. ACRONYMS AND ABBREVIATIONS

ALARA	as low as reasonably achievable
ANL	Argonne National Laboratory
ARMF	advanced reactivity measurement facility
ATM	approved testing materials
ATRC	advanced test reactor canal
BWR	boiling water reactor
CEQ	Council on Environmental Quality
CFR	Code of Federal Regulations
CFRMF	coupled fast reactivity measurement facility
DCG	Derived Concentration Guides
DFA	driver fuel assemblies
DOE	U.S. Department of Energy
EA	environmental assessment
ECF	Expended Core Facility
EIS	environmental impact statement
EPA	Environmental Protection Agency
EPCRA	Community Right-to-Know-Act
ERPG	Emergency Response Planning Guideline
ER&WM	environmental restoration and waste management
FAST	Flourinel and Storage Facility at INEL
FECF	fuel element cutting facility
FFTF	Fast Flux Test Facility
FSF	fuel storage facility
FSF	Underwater Fuel Storage Facility (located at INEL)
HLW	high-level waste

IDF	Inspection dose factor
IDLF	Immediately Dangerous to Life and Health Values
IDS	interim decay storage
IDLH	Immediately Dangerous to Life and Health Values
IEM	interim examination and maintenance
INEL	Idaho National Engineering Laboratory
IVS	in-vessel storage
LCF	latent cancer fatalities
LLW	low-level waste
MEPAS	Multimedia Environmental Pollutant Assessment System
MT	metric tons
MTHM	metric tons of heavy metal
MTR	materials test reactor
MTU	metric tons of uranium
NEPA	National Environmental Policy Act
NPDES	National Pollutant Discharge Elimination System
NRF	Naval Reactors Facility
NRHP	National Register of Historic Places
NTS	Nevada Test Site
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
PBF Canal	power burst facility canal
PEIS	programmatic environmental impact statement
PFP	Plutonium Finishing Plant
PSD	Prevention of Significant Deterioration
PUREX	Plutonium and Uranium Recovery through EXtraction

PWR	pressurized water reactor
RH-TRU	remote-handled transuranic material
RTEC	Registry of toxic effects for chemical substances
SBA	standard blanket assemblies
SHPO	Washington State Historic Preservation Officer
SNF	spent nuclear fuel
SPR	single-pass reactor
SRS	Savannah River Site
SS	single-shell tank
TDFA	test driver fuel assemblies
TEDF	Treated Effluent Disposal Facility
TFA	test fuel assemblies
TLV/TWA	Threshold Limit Values/Time Weighted Averages
TRIGA	Training, research, and isotope reactors built by General Atomic
WAC	Washington Administrative Code
WIPP	Waste Isolation Pilot Plant

ATTACHMENT A
FACILITY ACCIDENTS

Methods used to evaluate facility accidents associated with implementing the alternatives for SNF storage at Hanford are discussed in this attachment. The selection of radiological accidents for the analysis was based on information available in previously published safety or National Environmental Policy Act documents, as described in Section 5.15. Analyzed releases of nonradiological hazardous materials were based on actual or expected inventories at SNF management facilities using conservative release assumptions. Industrial construction and operational accidents are also evaluated based on the person-years needed to build and operate SNF facilities.

A.1 Radiological Accidents

The GENII computer code (Napier et al. 1988) was used to perform calculations for each facility to estimate the consequences of radionuclide releases to the atmosphere for onsite workers, members of the public at accessible locations on or near the site, individual residents at the site boundary, and the population within 80 km of the release location. Dose calculations used standard assumptions for the Hanford Site (Schreckhise et al. 1993), and health effects were estimated using recommendations of the International Commission on Radiological Protection in its Publication 60 (ICRP 1991). The risks of cancer and other long-term stochastic health effects as estimated by ICRP (1991) are based on populations exposed to relatively high doses of radiation at high dose rates. For estimating risk to populations where the total doses are below 20 rad, the ICRP recommended a low-dose reduction factor equal to 2. In this analysis, where accidents would yield individual dose estimates greater than 20 rad, the ICRP risk factors are used without the low dose correction to obtain the potential health effects.

Individual doses were estimated based on exposure of the receptor during the entire release, except where the release was sufficiently long that it could be divided into short-term and long-term components. In that case, onsite workers and members of the public at accessible onsite locations were assumed to remain in the path of the plume for the duration of the short-term component. The exposure duration for onsite individuals was assumed to be two hours, corresponding to the maximum time required to evacuate the Hanford Site in the event of an accident, and no ingestion pathways were considered. Offsite individuals were assumed to be exposed during the entire release, regardless of the accident duration. Because protective action guidelines specify mitigative actions to prevent consumption of contaminated food, the dose to offsite individuals and populations was estimated both with and without the food ingestion pathways. Reduced exposure to the plume or to contaminated ground surface as a result of early evacuation of offsite populations was not considered for the purposes of this analysis, although such action would certainly be taken in the event of a severe accident at the site.

Individual dose calculations were performed using atmospheric dispersion parameters that represented 95 percent conditions (i.e., the air concentrations used would not be exceeded more than 5 percent of the time). In the case of collective dose, the area surrounding the source was divided into 16 directions and 10 sectors by distance, and the dose was calculated for only the direction resulting in maximum collective exposure. Dose to the population was calculated using both 50 percent and 95 percent atmospheric dispersion parameters.

A.1.1 No Action Alternative

The No Action Alternative consists of fuel storage at existing Hanford facilities, including the 100-K Area wet storage basins; T Plant and a low-level burial ground in the 200-West Area; the 308, 324, 325, and 327 buildings in the 300 Area; and the Fast Flux Test Facility in the 400 Area. Maximum reasonably foreseeable accidents determined by previously published analyses were used for this evaluation, and the impacts of these accidents were reevaluated using a consistent set of parameters for the spectrum of receptors required for this document.

A.1.1.1 105-KE and 105-KW Basin Wet Storage. Airborne releases from the fuel storage pool are bounded by a postulated accident for the 105-KE and 105-KW Basins. In the accident, a cask is dropped and overturned in the fuel transfer area, with broken fuel elements spilling out of the cask, within the pool building, but away from the pool. The scenario assumes that the shipping cask ruptures, exposing all of the broken fuel elements in three canisters: 42 fuel elements each containing 22.5 kilograms (50 pounds) of fuel. The probability of this accident is estimated as 10^{-4} to 10^{-6} per year. The analysis assumes 10-year-old fuel-grade fuel (12 percent of plutonium content is plutonium-240). The source term is calculated by multiplying the inventory at risk by the release fraction. The calculation of the release fractions assumes the fuel heats but does not melt. Also, site evacuation is assumed, giving a two-hour time for calculation of the onsite release factor. The offsite release factor was calculated using an eight-hour release time. The calculated release quantity was 61 grams (0.14 pounds) for onsite exposure and 244 grams (0.54 pounds) for offsite exposure, resulting in the radionuclide releases listed in Table A-1. Recalculation of the doses for this analysis yields the results in Table A-2.

A cask drop involving broken fuel elements falling out of the cask would most likely be observed by the workers, who would also be alerted by area radiation alarms and the radiation monitor in attendance of a change in radiation intensity. The assumed 12 workers would likely be in Special Work Permit protective clothing, but typically would not be wearing respiratory

Table A-1. Estimated radionuclide releases for a dropped fuel cask accident in the 105-K wet storage basins.

	Release (Ci)	
Isotope	Onsite (2 hours)	Offsite (8 hours)
Yttrium-90	3.5 E-01	1.4 E+00
Strontium-90	3.5 E-01	1.4 E+00
Ruthenium-106	3.2 E-03	1.3 E-02
Antimony-125	7.3 E-03	2.9 E-02
Tellurium-125M	1.8 E-03	7.3 E-03
Cesium-134	7.9 E-03	3.2 E-02
Cesium-137	4.5 E-01	1.8 E+00
Cerium-144	1.7 E-03	6.8 E-03
Praseodymium-144	1.7 E-03	6.8 E-03
Praseodymium-144M	2.0 E-05	8.1 E-05
Promethium-147	1.2 E-01	4.9 E-01
Europium-154	5.4 E-03	2.1 E-02
Plutonium-236	1.3 E-08	5.4 E-08
Plutonium-238	2.9 E-03	1.2 E-02
Plutonium-239	6.7 E-03	2.7 E-02
Plutonium-240	3.5 E-03	1.4 E-02
Plutonium-241	2.7 E-01	1.1 E+00
Americium-241	5.7 E-03	2.3 E-02
Plutonium-242	1.3 E-06	5.1 E-06
Curium-244	2.8 E-04	1.1 E-03

Table A-2. Consequences of 105-KE Basin cask drop accident.

	Individual Impacts - Onsite and Offsite			
	Onsite Worker	Public Access Location	Individual Resident	
			All Pathways	Without Ingestion
Dose (rem)	3.4E+00	2.7E+00	a	5.2E-01
Fatal Cancer	1.4E-03	1.3E-03	a	2.6E-04

	Collective Impacts to Population within 80 km			
	50 percent E/Q[b]		95 percent E/Q	
	All Pathways	Without Ingestion	All Pathways	Without Ingestion
Dose (person-rem)	8.0E+02	3.5E+02	1.4E+04	6.1E+03
Fatal Cancers	4.0E-01	1.8E-01	6.9E+00	3.1E+00

a. The estimated potential dose to an offsite resident from the ingestion pathway is 1.4 rem. In practice, the dose would be limited by protective action guidelines that specify remedial measures if the potential dose is greater than 0.5 rem.
b. The term E/Q refers to the time - integrated air concentration at the receptor location for an acute release. It is analogous to the X/Q dispersion parameter used for a chronic release scenario.

protection. The workers would immediately evacuate the area to reduce their exposure to direct radiation (by increasing their distance from the source), for which their clothing provides no protection. Once at a distance, they would move upwind of the postulated airborne release before beginning decontamination procedures. Assuming the workers evacuate within 1 to 2 minutes, their dose would range from about 70 to 140 rem.[a] Using risk factors cited previously, the maximum probability of an individual contracting a fatal cancer from a dose of 140 rem would amount to about 0.06. The collective worker dose for such a scenario would amount to about 1800 person-rem for which one fatal cancer would be inferred. It should be noted, however, the risk factors used are not generally intended to be applied to large acute doses and such acute doses might produce minor near term adverse health effects.

Recent preliminary analyses, based on updated information on the ability of the 105-K Basins to withstand natural forces indicate that seismic-induced damage at the 105-K Basins could, under some circumstances, result in radiation exposure to the public and workers greater than that indicated in this EIS. The underlying concern is whether the fuel in its present

a. Acute doses of this magnitude are in the lower end of the range of doses that might produce symptoms of acute radiation syndrome in humans.

condition could become uncovered by loss of the basin water thereby resulting in larger releases of radionuclides to the atmosphere; in the present analysis the fuel is assumed to remain covered. A scenario in which the fuel would remain exposed to the air and allowed to burn is not considered a reasonably foreseeable accident for the time period covered by this EIS.

A.1.1.2 Liquid Release Scenario for 105-KE or 105-KW Basin. Accidental liquid releases from the 105-K Basins are bounded by seismic events or other mechanical disruption of the basin or its water supply system. The most probable scenario is a break in an 8-inch water supply line that overfills the storage pool causing water to overflow onto the surrounding soil (Bergsman 1995). The flow is assumed to continue for 8 hours before the supply is shut off, resulting in release of 2300 cubic meters (600,000 gallons) of water and 60% of the radionuclide inventory in the pool water. The inventory released from the 105-KE Basin is assumed to be 13 Ci tritium, 0.029 Ci cobalt-60, 9.2 Ci strontium-90, 0.042 Ci cesium-134, 12 Ci cesium-137/barium-137m, 0.0098 Ci plutonium-238, and 0.056 Ci plutonium-239.

The corresponding radionuclide inventory in the 105-KW Basin overflow pond is assumed to be as follows: 0.48 Ci tritium, 0.0013 Ci cobalt-60, 0.0031 Ci cesium-134, 0.22 Ci cesium-137, 1.1 Ci strontium-90, 5.9E-06 Ci plutonium-238, and 3.1E-05 Ci plutonium-239. The overflow is assumed to leach through the subsurface environment to the Columbia River. Because the transmission rate of the soil is estimated as 570 centimeters per day [based on DOE's Programmatic Environmental Impact Statement (PEIS) (Schramke 1993)], a leaching rate of 26.3 centimeters per day (10 inches per day) will not result in a ponded situation; therefore, the entire 2300 cubic meters (600,000 gal) of overflow will leach into the soil over an eight-hour period. Contaminants are assumed to travel through the vadose zone, through the saturated zone to the Columbia River and in the Columbia River to receptors downstream. The flow discharge in the Columbia River is assumed to be under low-flow conditions of 1000 cubic meters per second (36,000 cubic feet per second) (Whelan et al. 1987), which represents the most conservative case for maximizing surface water concentrations. As a conservative assumption, the removal of water from the Columbia River is assumed to be 100 meters (328 feet) downstream of the point of entry of the contaminant into the river. The assessment addressed recreational activities (e.g., boating, swimming, fishing) in the Columbia River and use of the water as a drinking-water supply and for bathing, irrigation, etc. The collective risk of fatal cancer from the spill at the 105-KW Basin was estimated as approximately 1.1×10^{-13} fatal cancers for the maximum pathway and radionuclide (ingestion of plutonium-239 in fish) at 2800 years. The cumulative risk from all radionuclides and pathways amounted to approximately 6 x

10^{-13} fatal cancers. The corresponding risks from a spill at the 105-KE Basin were 2 x 10^{-10} fatal cancers for the maximum nuclide and pathway (also from ingestion of plutonium-239 in fish), and about 6 x 10^{-10} fatal cancers for all radionuclides and pathways (Whelan et al. 1994).

The overflow scenario described in the previous paragraph has been extrapolated to include a larger release because of recent concerns about the effects of a seismic event severe enough to breach joints in the basin. A crack in the basin would potentially release all of the basin water and perhaps some of the sludge to the subsurface environment, where it would be available for leaching to groundwater and transport to the Columbia River. Because the liquid overflow scenario assumes release of over half of the basin water, the risk to a downstream individual from release of all the basin water would be less that twice that estimated for the overflow scenario. Radionuclides in the sludge would be much less mobile and would leach into groundwater slowly, providing time for remediation and mitigation measures as necessary. Even if significant quantities of sludge remained in the subsurface soil for an extended period prior to clean up, the risk to the downstream individuals and population would not likely be substantially higher than that estimated for the overflow scenario.

This accident would not likely present any hazard to workers at the basin because the scenario is liquid to ground to groundwater and on to the Columbia River and does not involve a source of exposure to the close-in workers.

A.1.1.3 308 Building. The maximum reasonably foreseeable accident for airborne releases related to fuel storage at the 308 Building is dropping a transfer basket while moving fuel from the reactor core to the storage pool (WHC 1990). It was conservatively estimated that 15 fuel elements would have their cladding damaged, resulting in the release of 100 percent of the krypton-85 to the environment in 5 minutes. The probability of this accident is estimated as 10^{-2} to 10^{-4} per year. In the original Safety Analysis Report, the resulting dose was estimated at 0.013 rem to the worker, 8.6 x 10^{-4} rem to the onsite individual, and 8.6 x 10^{-5} rem at the site boundary. Collective dose to the population was not reported in the SAR. The individual doses correspond to a probability of fatal cancer of 5.2E-06 per year for the worker, 4.3E-07 per year for the onsite member of the public, and 4.3E-08 per year at the site boundary.

This information is provided in more detail in WHC (1990), which, however, does not detail the total quantity of krypton-85 released in any of its accident scenarios. Because release quantities for krypton-85 were not available, the consequences of this accident were not reevaluated for this analysis. Note that the SAR worker evaluation is for an individual in the facility who is assumed to evacuate within 5 minutes. This is a somewhat different analysis from those for the other worker consequences presented for the Hanford Site, which assume a worker remains outside the facility at the point of maximum air concentration for a period of up to 2 hours.

A transfer basket drop that results in damage to 15 fuel elements would most likely be observed by the workers, who would also be alerted by area radiation alarms and the radiation monitor in attendance of a change in radiation intensity. The assumed 12 workers would likely be in Special Work Permit protective clothing, but typically would not be wearing respiratory protection. The workers would immediately evacuate the area to reduce their exposure to direct radiation (by increasing their distance from the source), for which their clothing provides no protection. Once at a distance, they would move upwind of the postulated airborne release before beginning decontamination procedures. It was estimated (WHC 1990) that the workers would receive a dose of 13 millirem. The collective worker dose would amount to about 0.2 person-rem, and no latent cancer fatalities would be predicted for these workers.

A.1.1.4 324 Building. The greatest potential safety concern at the 324 Building comes from a safety assessment of the current levels of potentially highly mobile radioactive material in B-Cell (PNL 1992a). The potential failure of the 324 Building exhaust ventilation system in a 0.1 g seismic event, along with shaking of highly mobile holdup material in the 324 Building hot cells, could cause a total release of 610 Ci of cesium-137 and 310 Ci of strontium-90 within 12 hours. Of this total, approximately 55 percent (340 Ci of cesium-137 and 170 Ci of strontium-90) would be released in the first two hours. The probability of the initiating seismic event is 4×10^{-4} per year, and the other events leading to the release are assumed in this analysis to occur with certainty. The consequences of this accident are presented in Table A-3. In comparison to this accident, other potential releases from the building are judged to be insignificant, or they have been determined to be less probable because of radioactive material containment or handling frequency. The consequences associated with this accident are a result of existing contamination in the 324 Building hot cells, and neither its likelihood nor its severity depend on the presence of spent fuel in the facility. The actual contribution of spent fuel to releases from the accident is assumed to be negligible compared with that of other sources.

A seismic event that causes the failure of the 324 Building exhaust ventilation system and releases significant quantities of non-spent nuclear fuel-related radioactive materials from the

Table A-3. Consequences of a seismic event at the 324 Building.

	Individual Impacts - Onsite and Offsite			
	Onsite Worker	Public Access Location	Individual Resident	
			All Pathways	Without Ingestion
Dose (rem)	1.1E+03	1.0E+02	a	3.5E+01
Fatal Cancer	a	1.0E-01	a	3.5E-02

	Collective Dose to Population within 80 km			
	50 percent E/Q[b]		95 percent E/Q	
	All Pathways	Without Ingestion	All Pathways	Without Ingestion
Dose (person-rem)	2.1E+05	1.8E+03	1.9E+06	1.6E+04
Fatal Cancers	1.0E+02	9.0E-01	9.7E+02	8.2E+00

a. These doses are sufficiently high that application of long-term risk factors is inappropriate. An acute total body dose of greater than 1,000 rem would be expected to be fatal from other mechanisms within a relatively short time. The estimated potential dose to an offsite resident from the ingestion pathway is 5.4E+03 rem. In practice, the dose would be limited by protective action guidelines, which specify remedial measures if the potential dose is greater than 0.5 rem.
b. The term E/Q refers to the time - integrated air concentration at the receptor location for an acute release. It is analogous to the X/Q dispersion parameter used for a chronic release scenario.

building could occur at any time, whether or not there were workers in the building. An earthquake of sufficient intensity to cause the ventilation failure would surely be noticed by any workers in the building. In all likelihood, area radiation alarms would also sound. The assumed 50 workers would immediately evacuate the building and move to a position upwind of the building. Although speculative, the workers might receive as much as 25 rem before reaching a completely safe zone. If that were the case, they would probably be restricted from further radiation worker pending results of reading their dosimeters and completion of a medical evaluation. The maximum probability of an individual contracting a fatal cancer from such a dose would amount to about 0.02. The postulated collective dose would amount to about 1300 person-rem, from which one latent cancer fatality might be inferred. Based only on the estimated initiating earthquake frequency, the chances of these consequences occurring would be about 1 in 5,000 per year.

A.1.1.5 325 Building. A severe earthquake, without subsequent fire, is the maximum reasonably foreseeable accident for the 325 Building (PNL 1992b). It is postulated that an earthquake would cause windows to break but not cause general or local structural collapse. Doors may be jammed open after building evacuation, leaving additional openings for unfiltered releases. Building power or ventilation could be lost. Further damage would be caused to glove boxes and the contents of shelves and cabinets. The expected effects are considered to be the most severe that could result from a 0.135 g horizontal acceleration, corresponding to the 2×10^{-4} per year seismic event for which protection is required by DOE design criteria for a new structure.

Radionuclide releases associated with this accident are listed in Table A-4. It should be noted that the environmental releases associated with the earthquake scenario are from all sources in the 325 Building; fuel storage activities account for only a small fraction of the total. Because these releases consist of a variety of chemical forms, the dose factors used for calculation of the consequences represented the maximum dose for all radionuclides in the total release. The consequences of this accident are presented in Table A-5.

An earthquake that results in openings for unfiltered releases from the 325 Building releasing significant quantities of non-spent nuclear fuel-related radioactive materials could occur at any time, whether or not there were workers in the building. An earthquake of sufficient intensity to cause damage to the ventilation system and possibly glove boxes and windows would surely be noticed by any workers in the building. Whether area radiation monitors alarmed or not, the assumed 50 workers would immediately evacuate the building and, once outside, would move to a position upwind of the building. Although speculative, the workers might receive as much as 3 rem before reaching a completely safe zone. The maximum probability of latent fatal cancer for such a dose would be 0.001. The postulated collective dose would amount to about 150 person-rem, from which no latent cancer fatalities would be inferred.

A.1.1.6 327 Building. The postulated maximum reasonably foreseeable accident for fuel storage at the 327 Building consists of mechanical damage to fuel pins and subsequent fire involving reactive fuel within a hot cell (WHC 1987). Because of the variety of activities that can occur in the hot cells, specific details of the accident were not postulated. The mechanical damage would breach the pin cladding and immediately release the gaseous fission products in the fuel-cladding gap. The subsequent fire would cause complete reaction of reactive fuel forms.

Table A-4. Radionuclide releases for the 325 Building earthquake scenario.[a]

Releases in the first 2 hours		
Radionuclide	Ci	
Tritium	0.0425	
Krypton-85	66.2	
Radionuclide	Ci as nitrate	Ci as oxide
Thorium-232	2.23E-10	2.32E-06
Uranium-238	1.04E-08	4.17E-05
Uranium-235	5.34E-10	1.16E-06
Uranium-233	1.36E-06	4.68E-07
Neptunium-237	6.88E-07	2.36E-07
Plutonium-238	0.002016	0.000772
Plutonium-239	0.002047	0.001203
Plutonium-240	0.001037	0.000609
Plutonium-241	0.051751	0.030407
Americium-241	0.000877	0.000343
Plutonium-242	2.88E-07	1.65E-07
Americium-243	2.09E-05	7.17E-06
Curium-244	0.003130	0.001075
Activity released after the first 2 hours but within the first 4 days		
Radionuclide	Ci as nitrate	Ci as oxide
Thorium-232	4.08E-10	2.01E-06
Uranium-238	1.91E-08	3.61E-05
Uranium-235	9.76E-10	1.0E-06
Uranium-233	7.08E-07	3.49E-07
Neptunium-237	3.58E-07	1.76E-07
Plutonium-238	0.002231	0.000614
Plutonium-239	0.008545	0.001143
Plutonium-240	0.004329	0.000579
Plutonium-241	0.216022	0.028896
Americium-241	0.001077	0.000276
Plutonium-242	1.41E-06	1.56E-07
Americium-243	1.08E-05	5.34E-06
Curium-244	0.001626	0.000801

a. Data from Draft Safety Analysis Report for the 325 Building (PNL 1992b).

Table A-5. Consequences of a seismic event at the 325 Building.

	Individual Impacts - Onsite and Offsite			
	Onsite Worker	Public Access Location	Individual Resident	
			All Pathways	Without Ingestion
Dose (rem)	1.3E+02	1.3E+01	a	5.9E+00
Fatal Cancer	1.0E-01	6.3E-03	a	3.0E-03

	Collective Dose to Population within 80 km			
	50 percent E/Q[b]		95 percent E/Q	
	All Pathways	Without Ingestion	All Pathways	Without Ingestion
Dose (person-rem)	4.5E+02	3.2E+02	4.1E+03	2.9E+03
Fatal Cancers	2.3E-01	1.6E-01	2.0E+00	1.5E+00

a. The estimated potential dose to an offsite resident from the ingestion pathway is 10 rem. In practice, the dose would be limited by protective action guidelines, which specify remedial measures if the potential dose is greater than 0.5 rem.
b. The term E/Q refers to the time - integrated air concentration at the receptor location for an acute release. It is analogous to the X/Q dispersion parameter used for a chronic release scenario.

Fission products are released to the environment through the ventilation system, which includes HEPA and activated charcoal filtration. The frequency of this accident is estimated as 10^{-4} to 10^{-6} per year. The hot cell inventory and the fraction of the inventory released are shown in Table A-6.

The previous analysis evaluated the most extreme case for damaged material containing the maximum allowable limits of fission products that had not been vented to release fission gases. In this case, fuel materials involved are assumed to be nonreactive in water and to contain a maximum fission product inventory of 6.5×10^6 Ci including 2500 Ci of halogens. Radionuclide releases from the fuel into the basin water and thence into the air above the water are based on U.S. Nuclear Regulatory Commission Regulatory Guide 1.25, which addresses accidents involving spent fuel in a storage pool. The consequences of the accident as evaluated for this document are listed in Table A-7.

Table A-6. Assumed inventories and release fractions for a 327 Building hot cell fire.

Radionuclide	Source Inventory (Ci)	Release Fraction	Radionuclide	Source Inventory (Ci)	Release Fraction
Halogens					
Iodine-129	3.16E-3	5E-03			
Noble Gases					
Krypton-85	4.63E+2	1	Xenon-131m	6.25E+2	1
Xenon-133m	8.55E+0	1	Xenon-133	1.03E+4	1
Volatile Solids					
Selenium-79	2.17E-2	2.5E-4	Rubidium-87	9.15E-7	2.5E-4
Cadmium-113m	9.02E+0	" "	Cadmium-115m	9.96E-6	" "
Cesium-134	1.37E+3	" "	Cesium-135	1.17E-1	" "
Cesium-137	7.27E+3	" "			
Nonvolatile Solids					
Strontium-89	4.41E-2	5E-6	Strontium-90	2.74E+3	5E-6
Yttrium-90	2.74E+3	" "	Yttrium-91	3.03E-1	" "
Zirconium-93	1.52E-1	" "	Zirconium-95	2.11E+0	" "
Niobium-95m	2.68E-2	" "	Niobium-95	4.55E+0	" "
Ruthenium-103	1.72E-3	" "	Ruthenium-106	1.34E+4	" "
Rhodium-103m	1.72E-3	" "	Rhodium-106	1.34E+4	" "
Palladium-107	1.60E-2	" "	Silver-110	1.94E+2	" "
Indium-114m	2.82E-8	" "	Indium-114	2.72E-8	" "
Tin-119m	2.58E+0	" "	Tin-121m	9.13E-2	" "
Tin-123	6.13E+0	" "	Tin-126	9.30E-3	" "
Antimony-124	5.96E-4	" "	Antimony-125	9.51E+2	" "
Antimony-126m	9.30E-3	" "	Antimony-126	1.30E-3	" "
Tellurium-123m	2.29E-3	" "	Tellurium-125m	2.32E+2	" "
Tellurium-127m	3.31E+0	" "	Tellurium-127	3.24E+0	" "
Tellurium-129m	2.55E-6	" "	Tellurium-129	1.62E-6	" "
Barium-137m	6.88E+3	" "	Cerium-141	2.23E-5	" "
Cerium-144	7.36E+3	" "	Praseodymium-144	7.36E+3	" "
Promethium-147	1.11E+4	" "	Promethium-148m	6.21E-5	" "
Promthium-148	4.28E-6	" "	Samarium-151	3.04E+2	" "
Europium-152	1.05E+0	" "	Europium-154	1.35E+2	" "
Europium-155	8.83E+2	" "	Gadolinium-153	1.24E-2	" "
Terbium-160	8.24E-3	" "	Holmium-166m	1.52E-3	" "
Heavy Metals					
Plutonium-239	2.24E+0	5E-6	Plutonium-240	2.21E+0	5E-6
Plutonium-241	3.46E+2	5E-6			

Table A-7. Consequences of 327 Building hot cell fire.

	Individual Impacts - Onsite and Offsite			
			Individual Resident	
	Onsite Worker	Public Access Location	All Pathways	Without Ingestion
Dose (rem)	2.2E-02	3.2E-02	a	2.3E-02
Fatal Cancer	8.6E-06	1.6E-05	a	1.1E-05

	Collective Dose to Population within 80 km			
	50 percent E/Q[b]		95 percent E/Q	
	All Pathways	Without Ingestion	All Pathways	Without Ingestion
Dose (person-rem)	4.7E+02	5.4E+00	4.3E+03	4.8E+01
Fatal Cancers	2.4E-01	2.7E-03	2.1E+00	2.4E-02

a. The estimated potential dose to an offsite resident from the ingestion pathway is 2.5 rem. In practice, the dose would be limited by protective action guidelines that specify remedial measures if the potential dose is greater than 0.5 rem.
b. The term E/Q refers to the time-integrated air concentration at the receptor location for an acute release. It is analogous to the X/Q dispersion parameter used for a chronic release scenario.

This accident involves mechanical damage to fuel pins, subsequent fire within a hot cell, and releases of radioactive material to the intact filtered ventilation system and on to the atmosphere. There would be no added source of radiation exposure to the close-in worker at the hot cell.

A.1.1.7 200-West Area Low-Level Waste Burial Grounds. The only accident postulated to have any significant radiological releases in the Burial Ground safety analysis report is briefly described as a vehicle impact on one or more EBR II casks followed by a fire (Saito 1992). Two vehicle impact scenarios were discussed in the document:

1. Severe impact or collision followed by a short-duration fire caused by a vehicular accident in the trench.

2. Extremely severe impact or collision followed by a long duration fire.

The consequences of the latter accident were evaluated for fuels containing maximum inventories of either fission product or transuranic radionuclides. The probability of the accident is estimated to be 9.8×10^{-6} per year. The consequences of the less severe accident

Table A-8. Radionuclide releases for spent nuclear fuel storage at 200-West Burial Ground, accident scenario 2 - extremely severe impact with long duration fire.

Radionuclide	Release (Ci) Maximum TRU Fuel[a]	Maximum FP Fuel[a]
Cobalt-60	1.4E-04	8.6E-04
Krypton-85	0.0E+00	2.4E+02
Strontium-90	5.6E-05	6.3E-03
Yttrium-90	5.6E-05	6.3E-03
Ruthenium-106	5.1E-04	6.6E-02
Cesium-137	7.2E-03	6.9E-01
Cerium-144	1.9E-04	1.9E-02
Praseodymium-144	1.9E-04	1.9E-02
Promethium-147	1.4E-04	1.5E-02
Europium-155	1.1E-05	1.3E-04
Uranium-233	0.0E+00	2.5E-07
Uranium-234	4.1E-05	3.5E-06
Uranium-236	0.0E+00	1.7E-09
Uranium-235	5.6E-10	5.3E-08
Uranium-238	2.0E-09	6.6E-10
Plutonium-238	7.5E-05	1.5E-05
Plutonium-239	1.4E-04	2.8E-05
Plutonium-240	4.0E-04	7.9E-05
Plutonium-241	2.3E-02	4.5E-04
Plutonium-242	1.4E-06	1.4E-08

a. Maximum TRU Fuel is that having the maximum concentration of transuranic radionuclides; maximum FP fuel has the maximum concentration of fission product radionuclides.

would be approximately an order of magnitude lower. The radionuclide releases for accident scenario 2 are shown in Table A-8; the accident consequences as re-evaluated for this document are presented in Table A-9. The maximum fission product inventory fuel yielded the highest consequences for offsite receptors where the ingestion pathway was considered. The maximum transuranic inventory was associated with higher consequences for the inhalation and external exposure pathways.

The severe impact or collision followed by fire as postulated here might have serious-to-fatal nonradiological consequences to drivers and passengers of the vehicles involved. It is assumed that two drivers and two passengers are involved. These individuals would evacuate

Table A-9. Consequences of cask impact accident and fire at 200-West Burial Ground.

	Individual Impacts - Onsite and Offsite			
			Individual Resident	
	Onsite Worker	Public Access Location	All Pathways	Without Ingestion
Maximum TRU[a]				
Dose (rem)	6.0E+00	6.7E-03	3.6E-03	2.5E-03
Fatal Cancer	2.4E-03	3.3E-06	1.8E-06	1.2E-06
Maximum FP[a]				
Dose (rem)	2.0E+00	2.2E-03	1.0E-01	9.5E-04
Fatal Cancer	8.0E-04	1.1E-06	5.0E-05	4.8E-07

	Collective Dose to Population within 80 km			
	50 percent E/Q[b]		95 percent E/Q	
	All Pathways	Without Ingestion	All Pathways	Without Ingestion
Maximum TRU[a]				
Dose (person-rem)	2.4E+00	1.8E+00	4.9E+01	3.6E+01
Fatal Cancers	1.2E-03	8.9E-04	2.4E-02	1.8E-02
Maximum FP[a]				
Dose (person-rem)	5.6E+01	6.6E-01	1.1E+03	1.3E+01
Fatal Cancer	2.8E-02	3.3E-04	5.7E-01	6.6E-03

a. Maximum TRU Fuel is that having the maximum concentration of transuranic radionuclides; maximum FP fuel has the maximum concentration of fission product radionuclides.
b. The term E/Q refers to the time-integrated air concentration at the receptor location for an acute release. It is analogous to the X/Q dispersion parameter used for a chronic release scenario.

the area, if they were able. Because it cannot be assured that after the collision either drivers or passengers would be able to evacuate the area to a safe distance from radiological consequences, the worst case is assumed, that the four individuals perish in this accident principally from trauma caused by the collision and fire. The likelihood of these consequences occurring are estimated at 1 chance in 100,000 per year.

A.1.1.8 T Plant. The maximum scenario for fuel storage at T Plant is a dropped fuel assembly inside the building (Jackson and Hanson 1978). The probability associated with this accident is estimated to be 2.8 x 10^{-3} per year. The release estimates assume damage to a fraction of the wafers in the dropped fuel module containing 4-year-cooled Shippingport PWR Core II fuel (a conservative assumption because the fuel has now been cooled for approximately 20 years). Other release assumptions include the following:

- 10% of nonvolatile radionuclides in broken fuel are released to the building floor
- 0.1% of the released particulate material is resuspended in the building
- All of the volatile krypton-85 is released to the building atmosphere
- Building filtration removed 98.6 percent of the particulate materials from the effluent exiting the stack.

Release estimates for this scenario are presented in Table A-10 and the consequences of the release are listed in Table A-11.

Because workers evacuate the canyon area when fuel assemblies are being moved to or from the casks or pool, there would be no opportunity for impacts on workers from a dropped fuel assembly in fuel storage at T Plant.

Table A-10. Releases for damaged assembly of Shippingport Core II fuel with 4-year decay at T Plant.

Radionuclide	Release (Ci)
Iron-55	5.0E-06
Cobalt-60	3.0E-06
Krypton-85	9.6E+00
Strontium-90	1.0E-04
Ruthenium-106	1.0E-04
Antimony-125	3.0E-06
Cesium-134	8.0E-06
Cesium-137	1.0E-04
Cerium-144	1.0E-04
Promethium-147	1.0E-04
Europium-154	3.0E-05
Plutonium-239	6.2E-07
Plutonium-240	1.6E-06
Plutonium-241	3.1E-04
Plutonium-242	3.9E-09

Table A-11. Consequences of fuel assembly damage at T Plant.

	Individual Impacts - Onsite and Offsite				
			Individual Resident		
	Onsite Worker	Public Access Location	Inhalation + External	Ingestion	Total
Dose (rem)	2.6E-04	5.7E-05	3.2E-05	5.3E-05	8.6E-05
Fatal Cancer	1.0E-07	2.8E-08	1.6E-08	2.6E-08	4.3E-08

	Collective Dose to Population within 80 km					
	50 percent E/Q[a]			95 percent E/Q		
	Inhalation + External	Ingestion	Total	Inhalation + External	Ingestion	Total
Dose (person-rem)	1.4E-02	1.6E-02	3.0E-02	3.2E-01	3.6E-01	6.8E-01
Fatal Cancers	7.2E-06	8.0E-06	1.5E-05	1.6E-04	1.8E-04	3.4E-04

a. The term E/Q refers to the time-integrated air concentration at the receptor location for an acute release. It is analogous to the X/Q dispersion parameter used for a chronic release scenario.

A.1.1.9 Fast Flux Test Facility (FFTF). The accident scenario for the handling and storage of irradiated FFTF fuel in the Fuel Storage Facility (FSF) is a liquid metal fire (Gantt 1989). The accident scenario is a spill of 11,793 kg of liquid sodium and subsequent fire. The spill is initiated by either an internal event or a seismic event that causes a break in the piping between the FSF and heat exchangers. The liquid sodium is assumed to ignite spontaneously and burn, releasing aerosols to the atmosphere. The probability of this accident is estimated to be 10^{-4} to 10^{-6} per year.

The radionuclide release is from cesium that has been leached from the fuel into the sodium. It is assumed for this accident that 0.1 percent of the elements are breached and that the sodium contains 0.9 μCi cesium-134 per gram of sodium and 5 μCi cesium-137 per gram of sodium. It is assumed that 35 percent of the sodium and cesium aerosols generated in the fire are released to the atmosphere. The total activity released is estimated as 3.7 Ci cesium-134 and 25 Ci cesium-137. The consequences of the accident as estimated are listed in Table A-12. Onsite individuals (workers and members of the public at onsite access locations) were assumed to be exposed during 0.4 percent of the total release, because the spilled sodium would require over 20 days to burn completely, and onsite individuals were assumed to be evacuated within 2 hours.

Table A-12. Consequences of liquid metal fire at the Fast Flux Test Facility.

	Individual Impacts - Onsite and Offsite				
	Onsite Worker	Public Access Location	Individual Resident		
			Inhalation + External	Ingestion	Total
Dose (rem)	7.3E-04	2.4E-04	1.6E-02	a	a
Fatal Cancer	2.9E-07	1.2E-07	7.9E-06	a	a

	Collective Dose to Population within 80 km					
	50 percent E/Q[b]			95 percent E/Q		
	Inhalation + External	Ingestion	Total	Inhalation + External	Ingestion	Total
Dose (person-rem)	2.6E+01	7.6E+03	7.6E+03	2.3E+02	6.4E+04	6.4E+04
Fatal Cancers	1.3E-02	3.8E+00	3.8E+00	1.2E-01	3.2E+01	3.2E+01

a. The estimated potential dose to an offsite resident including the ingestion pathway is 5.0 rem. In practice, the dose would be limited by protective action guidelines, which specify remedial measures if the potential dose is greater than 0.5 rem.
b. The term E/Q refers to the time-integrated air concentration at the receptor location for an acute release. It is analogous to the X/Q dispersion parameter used for a chronic release scenario.

An internal event or a seismic event that causes a break in the piping between the FSF and heat exchangers could occur whether workers were present or not. The event would surely be noticed by any workers in the building. In all likelihood, area radiation alarms would also sound. The assumed 50 workers would immediately evacuate the building and, once outside, would move to a position upwind of the building. Because this is an accident that involves a slow release of material to the atmosphere, it is speculated that dose to the close-in workers would not exceed 0.1 rem from this accident. The postulated collective dose would amount to about 5 person-rem, from which no latent cancer fatalities would be expected.

A.1.2 Decentralization Alternative

The Decentralization Alternative involves construction of several new facilities at Hanford, including new dry storage for spent fuel or a combination of new wet and dry storage. Options are also included for several types of fuel processing prior to storage. The consequences of new facilities are based on previously evaluated accidents for similar installations, adapted for the conditions and location of these facilities as assumed in this analysis.

A.1.2.1 New Wet Storage. This accident scenario is the same as that described for a dropped fuel container at the 100-K Basins. The releases are assumed to be the same as for the accident previously described (see Table A-1), but the evaluation was repeated for potential location of the new facility adjacent to the 200-East Area. The accident frequency in the No Action Alternative is also assumed for this alternative because the quantity of fuel handled in either case would be the same. The consequences of this accident for a new facility are shown in Table A-13.

A maximum reasonably foreseeable liquid release scenario has been postulated for the new pool storage facility for wet storage of nuclear fuels. The leak is based on a 20-cm (8-inch) water-supply pipe breaking inside of the pool building and releasing 7600 liters per minute (2000 gallons per minute). The flow is not shut off for 8 hours, resulting in 3600 cubic meters (960,000 gal) being added to the pool. Because the pool cannot handle this amount of liquid, there is an overflow of 2300 cubic meters (600,000 gal) in this 8-hour period. Because the transmission rate of the soil is estimated as 570 centimeters per day (220 inches per day) [based on DOE's Programmatic Environmental Impact Statement (PEIS) (Schramke 1993)], a leaching rate of 26.3 centimeters per day (10 inches per day) will not result in ponding; therefore, the entire volume of overflow will leach into the soil over an 8-hour period. The basin overflow does contain 61 percent of the basin-water radionuclide inventory, which is estimated as 1.8 Ci. The specific radionuclide inventory in the overflow pond is assumed to be as follows: 0.48 Ci tritium, 0.0013 Ci cobalt-60, 0.031 Ci cesium-134, 0.22 Ci cesium-137, 1.1 Ci strontium-90, 5.9E-06 Ci plutonium-238, and 3.1E-05 Ci plutonium-239. All of the constituents in this assessment are radionuclides. Contaminant migration is through the vadose zone, through the saturated zone to the Columbia River, and in the Columbia River to receptors downstream. The flow discharge in the Columbia River is assumed to be under low-flow conditions of 1000 cubic meters per second (36,000 cubic feet per second) (Whelan et al. 1987), which represents the most conservative case for maximizing surface water concentrations. As a conservative assumption, the removal of water from the Columbia River is assumed to be 100 meters (328 feet) downstream of the point of entry of the contaminant into the river. The assessment addressed recreational activities (e.g., boating, swimming, fishing) in the Columbia River and use of the water as a drinking-water supply and for bathing, irrigation, etc. The overall risk of fatal cancer from this accident was found to be less than 10 chances in a billion. (Whelan et al. 1994).

Table A-13. Consequences of cask drop accident at new wet storage facility adjacent to the 200-East Area.

	Individual Impacts - Onsite and Offsite			
			Individual Resident	
	Onsite Worker	Public Access Location	All Pathways	Without Ingestion
Dose (rem)	8.7E-01	6.3E-02	3.6E-01	1.3E-01
Fatal Cancer	3.5E-04	3.1E-05	1.8E-04	6.4E-05
	Collective Dose to Population within 80 km			
	50 percent E/Q[a]		95 percent E/Q	
	All Pathways	Without Ingestion	All Pathways	Without Ingestion
Dose (person-rem)	3.7E+02	1.7E+02	6.0E+03	2.7E+03
Fatal Cancers	1.9E-01	8.4E-02	3.0E+00	1.3E+00

a. The term E/Q refers to the time-integrated air concentration at the receptor location for an acute release. It is analogous to the X/Q dispersion parameter used for a chronic release scenario.

A cask drop involving broken fuel elements falling out of the cask at a new wet storage facility would be the same as discussed in Section A.1.1.1. No prompt radiation illness or latent cancer fatalities would be predicted for workers in this scenario.

The accident scenario at the 105-KE and 105-KW Basins and its results described under the No Action Alternative would also be applicable under the Decentralization Alternative prior to transport of fuel to a new storage facility.

A. 1.2.2 New Dry Storage - Small Vault or Cask Facility. The maximum reasonably foreseeable accident for the dry storage facility is assumed to be the same as that for a previously evaluated accident involving transport of FFTF fuel (DOE 1986b). This accident is used as a surrogate for a dry storage facility accident involving an impact by either an internal or external initiator that results in a fire. The release associated with this accident is estimated at 5.4E+02 Ci, based on the hypothetical scenario of six FFTF fuel assemblies irradiated to 150 MWD/Kg being subjected to a severe impact followed by a fire. The fuel pins rupture on impact or on heating in the fire, which burns for an hour before being extinguished. The probability of such an accident resulting in breach of the transport cask is estimated to be 9×10^{-7} or lower for 100 onsite shipments of FFTF fuel. The estimated frequency for this accident in the Decentralization Alternative has been adjusted to 6×10^{-6} per year based on the quantity of fuel that would be handled in loading the dry storage facility. Volatiles, particulates, and noble gases are released to the atmosphere. The estimated radionuclide releases are listed in Table A-14, and the radiological consequences are presented in Table A-15.

Table A-14. Estimated radionuclide releases for cask impact accident and fire at new dry storage facility, based on FFTF fuel transport.

Radionuclide	Release (Ci)
Tritium	4.6 E+01
Krypton-85	4.0 E+02
Strontium-90	2.7 E-02
Ruthenium-106	1.3 E+00
Cesium-134	1.7 E+01
Cesium-137	8.0 E+01
Plutonium-238	8.9 E-04
Plutonium-239	1.6 E-03
Plutonium-240	1.8 E-03
Plutonium-241	7.3 E-02
Americium-241	1.0 E-03

Table A-15. Consequences of cask impact accident with fire at new dry storage facility.

	Individual Impacts - Onsite and Offsite			
			Individual Resident	
	Onsite Worker	Public Access Location	All Pathways	Without Ingestion
Dose (rem)	1.2E+02	7.6E-02	a	5.0E-02
Fatal Cancer	9.4E-02	3.8E-05	a	2.5E-05
	Collective Dose to Population within 80 km			
	50 percent E/Q[b]		95 percent E/Q	
	All Pathways	Without Ingestion	All Pathways	Without Ingestion
Dose (person-rem)	8.0E+03	4.5E+01	1.6E+05	9.0E+02
Fatal Cancers	4.0E+00	2.3E-02	8.1E+01	4.5E-01

a. The estimated potential dose to an offsite resident from the ingestion pathway is 10 rem. In practice, the dose would be limited by protective action guidelines, which specify remedial measures if the potential dose is greater than 0.5 rem.
b. The term E/Q refers to the time-integrated air concentration at the receptor location for an acute release. It is analogous to the X/Q dispersion parameter used for a chronic release scenario.

An internal or external initiator that causes a breach followed by fire in a dry storage facility would surely be noticed by nearby workers. In all likelihood, area radiation alarms would also sound. The assumed 12 workers would immediately evacuate the area and, once at a safe distance, would move to a position upwind of the building. Evacuation time to that location would be measured in minutes. The dose to close-in workers is speculated to be about 3 rem. The maximum probability of latent fatal cancer from such a dose would be 0.001. The postulated collective dose would amount to about 36 person-rem, from which no latent cancer fatalities would be expected.

A.1.2.3 New Fuel Stabilization Facility. The maximum reasonably foreseeable radiological accident for fuel processing (either calcine or solvent extraction) is a uranium metal fire in a storage vessel (DOE 1986b; Bergsman 1995). The frequency of this accident is estimated at 10^{-4} to 10^{-6} per year. Releases for the accident from a new facility adjacent to the 200-East Area are listed in Table A-16. The total release assumes that fuel burns for a period of 20 hours; therefore, doses to onsite receptors were calculated on the basis that they were exposed for 2 hours (or 10 percent of the total release, assuming a constant release rate for the duration of the fire). The consequences of the accident are listed in Table A-17.

This accident involves a uranium fire in a storage vessel with releases of radioactive material to the atmosphere. There would be no added source of radiation exposure of the close-in worker in the processing facility.

A.1.3 1992/1993 Planning Basis Alternative

Accidents and consequences would be essentially the same as those for the Decentralization Alternative.

A.1.4 Regionalization Alternative

Accidents and consequences would be essentially the same as for the Decentralization Alternative. The accident frequencies for a cask impact and fire at handling and storage facilities were adjusted to account for the quantity of imported or exported fuel handled in each of the suboptions at a receiving and canning facility or in loading storage facilities. For

Table A-16. Estimated airborne radionuclide release from shear/leach/calcine stabilization facility as a result of maximum reasonably foreseeable accident (uranium metal fire in storage vessel).

Radionuclide	Previous Estimate of Available Material (Ci)[a]	Current Estimate of Available Material (Ci)[b]	Release Fraction	Total Curies Released
Tritium	3.20E+02	2.16E+02	1E+00	2.16E+02
Carbon-14	2.60E-01	7.84E-05	1E+00	7.84E-05
Krypton-85	6.50E+03	4.12E+03	1E+00	4.12E+03
Strontium-89	1.90E+05	4.27E-16	1E-07	N/A[c]
Strontium-90	5.10E+04	4.76E+04	1E-07	4.76E-03
Yttrium-91	3.30E+05	5.03E-13	1E-07	N/A
Zirconium-95	4.80E+05	2.44E-11	1E-07	N/A
Ruthenium-103	1.20E+05	3.00E-22	1E-06	N/A
Ruthenium-106	2.50E+05	3.89E+02	1E-06	3.89E-04
Antimony-125	9.40E+03	8.82E+02	1E-07	8.82E-05
Tellurium-127m	6.90E+03	1.79E-06	1E-06	N/A
Tellurium-129m	2.30E+03	1.85E-28	1E-06	N/A
Iodine-129	1.90E-02	2.00E-02	1E-02	2.00E-04
Iodine-131	4.10E-01	0.00	1E-02	0.00
Cesium-134	2.20E+04	1.04E+03	1E-06	1.04E-03
Cesium-137	6.40E+04	5.87E+04	1E-06	5.87E-02
Cerium-141	7.80E+04	6.01E-28	1E-07	N/A
Cerium-144	9.30E+05	2.27E+02	1E-07	2.27E-05
Promethium-147	1.70E+05	1.57E+04	1E-07	1.57E-03
Plutonium-238	2.50E+02	3.54E+02	1E-07	3.54E-05
Plutonium-239	7.70E+02	7.70E+02	1E-07	7.70E-05
Plutonium-240	4.10E+02	4.18E+02	1E-07	4.18E-05
Plutonium-241	4.90E+04	3.13E+04	1E-07	3.13E-03
Americium-241	5.60E+01	6.53E+02	1E-07	6.53E-05
TOTAL				4.34E+03

a. Mixed (80 percent Mark IV, 20 percent Mark IA) N-fuel irradiated to 3,000 MWD/MTU, cooled 180 days after discharge from reactor. Estimated 7 MTU uranium metal fuel burned and radionuclides released in 20 hours.
b. Mark IA N-fuel (100 percent) irradiated to 3,000 MWD/MTU, cooled 10 years after discharge from reactor. Estimated 7 MTU uranium metal fuel burned and radionuclides released in 20 hours.
c. N/A = Not applicable.

Table A-17. Consequences of uranium metal fire at fuel stabilization facility.

	Individual Impacts - Onsite and Offsite			
			Individual Resident	
	Onsite Worker	Public Access Location	All Pathways	Without Ingestion
Dose (rem)	2.1E-04	4.4E-05	6.9E-03	2.7E-04
Fatal Cancer	8.3E-08	2.2E-08	3.4E-06	1.3E-07

	Collective Dose to Population within 80 km			
	50 percent E/Q[a]		95 percent E/Q	
	All Pathways	Without Ingestion	All Pathways	Without Ingestion
Dose (person-rem)	9.1E+00	5.3E-01	1.3E+02	7.3E+00
Fatal Cancers	4.6E-03	2.6E-04	6.4E-02	3.6E-03

a. The term E/Q refers to the time-integrated air concentration at the receptor location for an acute release. It is analogous to the X/Q dispersion parameter used for a chronic release scenario.

Regionalization A (all fuel except defense fuel would be shipped offsite) the frequency was assumed to be the same as in Decentralization (6E-06 per year). The frequency in Regionalization B (Western fuel comes to Hanford) is slightly higher (7E-06) because of the additional fuel that would be handled. The Regionalization C Alternative is assigned a lower frequency (5E-06) when all SNF is shipped offsite.

A.1.5 Centralization Alternative

The Centralization Alternative consists of two options at Hanford - a minimum option in which all DOE spent fuel at Hanford is transported offsite to another location for interim storage, and a maximum alternative that would result in storage of all DOE spent fuel at Hanford. Accident scenarios for the minimum option would include those discussed under the No Action Alternative prior to shipment of the fuel offsite. In addition, N reactor and SPR fuel would be stabilized prior to shipment in a facility similar to the shear/leach/calcine facility discussed under the Decentralization Alternative. The uranium metal fire accident discussed under that alternative is assumed to be the maximum reasonably foreseeable accident for a stabilization facility in this case as well. The estimated frequency for the cask impact and fire at storage or canning and shipping facilities has been adjusted to 5 x 10^{-6} per year based on the quantity of fuel that would be handled in the centralization minimum alternative.

The maximum option contains suboptions for wet or dry fuel storage with processing similar to those for the Decentralization Alternative, and the consequences are expected to be essentially the same as those described previously. The estimated frequency for the cask impact and fire at a receiving and canning or dry storage facility has been adjusted to 8 x 10^{-6} per year based on the quantity of imported fuel that would be handled in the Centralization Alternative, maximum option. The only additional installation that would be included in this option is the Expended Core Facility (ECF), which would be relocated from the INEL. The consequences of accidents at this facility are discussed in Volume 1, Appendix D of this document. It should be noted that the accident evaluation for the ECF at Hanford in Appendix D uses assumptions that are different from those used for the Hanford accidents in this attachment and therefore the risks associated with the ECF at Hanford cannot be compared directly with those for the other Hanford facilities presented here. The consequences of the ECF accidents using Hanford Site assumptions would be higher than those presented in Appendix D.

A.2 Nonradiological Accidents

For purposes of the analysis, a worst-case accident scenario was developed for each existing and planned facility. The details of the nonradiological accident scenario are presented in this section. The scenario involves a chemical spill within a building, followed by an environmental release from the normal exhaust system. It is assumed that the building remains intact but containment measures fail, allowing release to occur through the ventilation system. It is assumed that all, or a portion of, the entire inventory of toxic chemicals stored in each building is released. The environmental releases are modeled and the hypothetical concentrations at three receptor locations are compared to toxicological limits.

A.2.1 Chemical Lists

Chemical inventory and chemical emissions lists have been developed provided by alternative and facility (Bergsman 1995). These chemical lists are of three basic types. The first type is a "worst-case chemical inventory," prepared to comply with the Emergency Planning and Community Right-To-Know Act reporting requirement. For facilities that store SNF, this lists which ones are of particular interest. The second type, presented in the Facility Costs section, is a general statement listing proposed process chemicals. The third type of list is an estimate of proposed liquid effluents and airborne emissions, presented in the Facility Discharges section. Effluent and emissions data are not presented for every option.

A.2.2 Baseline Chemical Inventory Based on Existing Facilities

A baseline inventory of chemicals kept in SNF facilities was developed from chemical inventories for these facilities that were compiled to comply with the Emergency Planning and Community Right-To-Know Act. The existing storage facilities are 105-KE Basin, 105-KW Basin, PUREX (202A), T Plant (221T), 2736-ZB Building, 200W low-level burial grounds, Fast Fuel Test Facility (FFTF) (403 Building), 308 Building, 324 Building, 325 A&B Building, and 327 Building. The Emergency Planning and Community Right-To-Know Act lists used are from 1992.

Because most facilities have various missions, the need for an inventory of chemicals at these facilities may not be related to the storage of SNF. The assumption is made that the existing inventories represent the amounts and types of chemicals that may be needed in the future.

Table A-18 lists chemicals by facility, the regulated reportable quantity (RQ) in the event of an environmental release, the maximum quantity stored, its physical state (gas, solid, liquid), the reference where the chemical is listed, the hypothetical release fraction (1 for gases, 0.1 for liquids, and 0.01 for solids), the calculated total hypothetical chemical release, and the chemical's probable use.

In the table, a solid frame around a number indicates that a stored quantity exceeds the reportable quantity for that chemical; a double-lined frame indicates that a conservative hypothetical accidental release would exceed the reportable quantity. A total of seventeen chemicals fall in the latter category and have the highest probability to be released to the air. These seventeen chemicals are the ones that would demand the highest attention in an emergency plan.

Because a reportable quantity has not been defined for every chemical, the inherent toxicity of each chemical was also considered in assessing its importance. The release fractions used in the accidental spill scenario are conservative, higher than those reported in the literature by as much as three orders of magnitude (Hickey et al. 1991).

A.2.3 Proposed Facilities

Table A-19 is primarily derived from the Facility Costs section of the engineering design data (Bergsman 1995). However, the 105-KE Basin is used as a surrogate for a baseline chemical inventory for the wet storage facility because the Facility Cost section lists only sodium hydroxide and sulfuric acid.

Table A-19 lists chemicals by facility, the regulated reportable quantity (RQ) in the event of an environmental release, the maximum quantity stored, its physical state (gas, solid, liquid), the reference where the chemical is listed, the hypothetical release fraction (1 for gases, 0.1 for liquids, and 0.01 for solids), the calculated total hypothetical chemical release, and the chemical's probable use. In the table, a solid frame around a number indicates that a stored quantity exceeds the reportable quantity for that chemical; a double-lined frame indicates that a conservative hypothetical accidental release would exceed the reportable quantity. A total of six chemicals fall in the latter category and have the highest probability to be released to the air. These six chemicals are the ones that would demand the highest attention in an emergency plan.

A.2.4 Atmospheric Modeling

Effects to onsite workers, the nearest point of pubic access, and the public at the nearest offsite residence were estimated using the computer model EPIcode (DOE 1993b). EPIcode uses a straight line Gaussian plume model and characteristics of an individual chemical to estimate downwind concentrations independent of direction. The 95 percent meteorological parameters were used to determine the wind speeds and stability class used for the simulation. In each case, stability class F was used. Wind speeds of 0.89 meters per second (2.0 miles per hour) were used for calculating effects to an onsite worker, the nearest point of public access, and at the nearest offsite residence. Other criteria used in the model simulations can be found in DOE (1993a).

Table A-18. Baseline Chemical Inventory for Existing Facilities in SNF Storage Locations

Facility/Chemical Name	RQ[a] lb.	Maximum Quantity Stored	Units[b]	Physical State[c]	Ref.[3]	Release Fraction[e]	Total Release	Comments
105-KE								
argon	na	42	lb.	g	1	1	42	Used to create an inert atmosphere.
chlorine	10	**9000**	lb.	s	2	0.01	***90***	For treatment of intake water.
EDTA disodium salt	na	267.5	lb.	s	2,1	0.01	2.675	Used for water analysis.
hydrogen peroxide	na	7	lb.	l	2,1	0.1	0.7	Cleaning and disinfection.
methane	na	42	lb.	g	1	1	42	Fuel.
nitrogen	na	6	lb	g	1	1	6	Used to create an inert atmosphere.
paraffin	na	1485	gal	l	1	0.1	148.5	Shielding and insulation.
PCB	1	**4701**	lb.	l	1	0.1	***470.1***	Transformer coolant.
potassium permanganate	100	8.8	lb.	s	2,1	0.01	0.088	Reagent.
sodium carbonate	na	2.2	lb.	s	2,1	0.01	0.022	Reagent and cleaner.
sodium hydroxide	1000	**3000**	gal	l	2	0.1	300	For water pH control.
sodium metabisulfite	na	4	lb.	s	2,1	0.01	0.04	Neutralizer.
stannous chloride	na	133	lb.	s	2,1	0.01	1.33	Reagent, catalyst, and cleaner.
sulfuric acid	1000	**3000**	gal	l	2	0.1	300	For water pH control.
105-KW								
argon	na	16	lb.	g	1	1	16	Used to create an inert atmosphere.
chlorine	10	**9000**	lb.	s	2	0.01	***90***	For treatment of intake water.
EDTA disodium salt	na	267.5	lb.	s	2	0.01	2.675	Used for water analysis.
ethylene glycol	1	**507.4**	lb.	l	3	0.1	***50.74***	Antifreeze.
helium	na	2	lb.	g	1	1	2	Used to create an inert atmosphere.
hydrogen peroxide	na	1.74	lb.	l	2,1	0.1	0.174	Cleaning and disinfection.
kerosene	10	**385**	gal.	l	1	0.1	***38.5***	Fuel.
lubricating oil	na	275	gal.	l	1	0.1	27.5	Equipment lubrication.
methane	na	288	lb.	g	1	1	288	Fuel.
nitrogen	na	48	lb.	g	1	1	48	Used to create a nonflammable atmosphere.
polyacrylamide	1	**110**	gal.	l	1	0.1	***11***	Vinyl polymer.
potassium permanganate	100	8.8	lb.	s	2,1	0.01	0.088	Reagent.
sodium carbonate	na	2.2	lb.	s	2	0.01	0.022	Reagent, cleaner.
sodium hydroxide	1000	**3000**	gal	l	2	0.1	300	For water pH control.
sodium metabisulfite	na	1500	gal.	s	2	0.01	15	Neutralizer.
stannous chloride	na	133	lb.	s	2	0.01	1.33	Reagent, catalyst, and cleaner.
sulfuric acid	1000	**3000**	gal	l	2	0.1	300	For water pH control.
PUREX (202A)								
bromochlorodifluoromethane	na	308	lb.	g	1	1	308	Halon fire extinguishers.
bromotrifluoromethane	na	1800	lb.	g	1	1	1800	Halon fire extinguishers.
cadmium nitrate tetrahydrate	1	**1488**	lb.	l	1	0.1	***148.8***	Use unknown.
diesel fuel	10	**10700**	gal	l	1	0.1	***1070***	Fuel.
EDTA disodium salt	na	4	lb.	s	1	0.01	0.04	Used for chemical analysis.

Table A.18 (contd)

Facility/Chemical Name	RQ[a] lb.	Maximum Quantity Stored	Units[b]	Physical State[c]	Ref.[3]	Release Fraction[e]	Total Release	Comments
ferris sulfamate	na	320	lb.	l	1	0.1	32	Use unknown.
mercury	1	34	lb.	l	1	0.1	3.4	Use unknown.
methanol	1	10	lb.	l	1	0.1	1	Fuel.
mineral oil	na	2178	lb.	l	1	0.1	217.8	Coolant and equipment lubricant.
nitrogen	na	2520	cu ft	g	1	1	2520	Used to create a nonflammable atmosphere.
PCB	1	1	lb.	l	1	0.1	0.1	Transformer coolant.
potassium permanganate	100	3	lb.	s	1	0.01	0.03	Reagent.
sodium fluoride	1000	10	lb.	s	1	0.01	0.1	Use unknown.
sodium hydroxide	1000	1843	lb.	l	1	0.1	184.3	For water pH control.
sodium metabisulfite	na	6	lb.	s	1	0.01	0.06	Neutralizer.
sodium nitrite	100	2008	lb.	l	1	0.1	200.8	Reagent.
sufuric acid	1000	200	lb.	l	1	0.1	20	Battery acid.
T-Plant (221T)								
argon	na	940	cu ft	g	1	1	940	Used to create an inert atmosphere.
helium	na	200	cu ft	g	1	1	200	Used to create an inert atmosphere.
methane	na	20000	cu ft	g	1	1	20000	Fuel.
nitrogen	na	200	cu ft	g	1	1	200	Used to create a nonflammable atmosphere.
oxalic acid	na	405	lb.	s	1	0.01	4.05	Reagent.
phosphoric acid	5000	372	lb.	l	1	0.1	37.2	Reagent, catalyst.
potassium permanganate	100	220	lb.	s	1	0.01	2.2	Reagent.
propane	na	1020	lb.	g	1	1	1020	Fuel.
sodium	10	1800	lb.	s	1	0.01	18	Industrial coolant.
sodium hydroxide	1000	7600	lb.	s	1	0.01	76	For water pH control and as reagent.
sodium nitrite	100	800	lb.	s	1	0.01	8	Reagent.
2736-ZB Bldg								
commercial adhesive	na	6	lb.	l	1	0.1	0.6	Super 77 adhesive.
commercial cleaners	na	87	lb.	l, s	1	0.1	8.7	Comet, 409, Lecta clean, 3c's window cleaner.
commercial lubricant	na	10	lb.	l	1	0.1	1	WD40.
LL- burial Grounds								
no chemical inventory noted								
FFTF (403 Bldg.)								
argon	na	3500	lb.	l	1	0.1	350	Used to create an inert atmosphere.
argon	na	880	cu ft	g	1	1	880	Used to create an inert atmosphere.
bromotrifluoromethane	na	160	lb.	g	1	1	160	Halon fire extinguishers.
helium	na	1000	cu ft	g	1	1	1000	Used to create an inert atmosphere.
sodium	10	240000	lb.	l	1	0.1	24000	Industrial coolant.
sodium potassium alloy	10	2780	lb.	l	1	0.1	278	Industrial coolant.
sulfuric acid	1000	12	lb.	l	1	0.1	1.2	Reagent.

Table A.18 (contd)

Facility/Chemical Name	RQ[a] lb.	Maximum Quantity Stored	Units[b]	Physical State[c]	Ref.[3]	Release Fraction[c]	Total Release	Comments
308 Bldg								
1,2-ethanediol	na	18832	lb.	l	2	0.1	1883.2	Use unknown.
acetone	1	1	gal	l	1	0.1	0.1	Solvent.
acetylene	na	20	lb.	g	1,2	1	20	Welding.
argon	na	832	lb.	g	1,2	1	832	Used to create an inert atmosphere.
bromotrifluoromethane	na	95	lb.	g	1,2	1	95	Halon fire extinguishers.
chem reagents, wet lab	--	>4	lb.	mixed	1			Assorted laboratory reagents in small quantities.
EDTA disodium salt	na	4	lb.	s	2	0.01	0.04	Used for chemical analysis.
ethyl alcohol	na	48	lb.	l	1,2	0.1	4.8	Solvent.
ethylene glycol	1	**2015**	gal	l	1	0.1	***201.5***	Antifreeze.
glycerine	na	1	lb.	l	1	0.1	0.1	Reagent.
heat transfer oil	na	235	gal	l	1,2	0.1	23.5	Coolant.
helium	na	408	lb.	g	1,2	1	408	Used to create an inert atmosphere.
hydrogen/argon mix	na	598	lb.	g	1	1	598	Use unknown.
hydroquinone	na	45	gal	l	1	0.1	4.5	Use unknown.
liquid nitrogen	na	62275	gal	l	1	0.1	6227.5	Nonflammable coolant.
methane/argon mix	na	104	lb.	g	1,2	1	104	Use unknown.
mineral oil	na	235	gal	l	1	0.1	23.5	Coolant and equipment lubricant.
nitrogen	na	419942	lb.	g	1,2	1	419942	Used to create a nonflammable atmosphere.
oxygen	na	20	lb.	g	2	1	20	Welding.
potassium permanganate	100	2	lb.	g	2	1	2	Reagent.
sodium bisulfite	5000	2	lb.	s	2	0.01	0.02	Used for chemical analysis.
stoddard solvent	na	11	lb.	l	1	0.1	1.1	WD40.
sulfur hexafluoride	na	539	lb.	l,g	1	0.1	53.9	Electrical system.
sulfuric acid	1000	157	gal	l	1	0.1	15.7	Reagent.
tergitol	na	41	lb.	l	2	0.1	4.1	Detergent and surfactant, nonoxynol.
x-ray film (Ag)	1	**2710**	lb.	s	1	0.01	***27.1***	Photographic plates.
324 Bldg								
acetylene	na	690	cu ft	g	1,2	1	690	Welding.
alkyl dimethyl benzyl ammonium	5000	5	gal	l	1,2	0.1	***0.5***	Degreaser, Dearcide 717 (14-200).
argon	na	1250	cu ft	g	1,2	1	1250	Used to create an inert atmosphere.
bis-tri-n-butyltin oxide	na	5	gal	l	1,2	0.1	***0.5***	Degreaser, Dearcide 717 (14-200).
carbon dioxide	na	250	lb.	g	1,2	1	250	Use unknown.
helium	na	213	cu ft	g	1,2	1	213	Used to create an inert atmosphere.
nitrogen	na	456	cu ft	g	1,2	1	456	Used to create a nonflammable atmosphere.
oxygen	na	620	cu ft	g	1,2	1	620	Welding.
poly oedmi ethylene dichloride	1	**16**	gal	l	1,2	0.1	***1.6***	Use unknown, Dearcide 722 (14-730)
potassium hydroxide	1000	18	gal	l	1,2	0.1	1.8	Use unknown, Dearborn 727 (1-688)

Table A.18 (contd)

Facility/Chemical Name	RQ[a] lb.	Maximum Quantity Stored	Units[b]	Physical State[c]	Ref.[3]	Release Fraction[e]	Total Release	Comments
325 Bldg								
acetylene	na	360	cu ft	g	1,2	1	360	Welding.
aluminum	na	10	lb.	s	1,2	0.01	0.1	Reagent.
aluminum oxide	na	24	lb.	s	1,2	0.01	0.24	Reagent.
aluminum sulfate dihydrate	5000	11	lb.	s	2	0.01	0.11	Reagent.
ammonium bicarbonate	5000	50	lb.	s	1,2	0.01	0.5	Reagent.
ammonium nitrate	na	23	lb.	s	1,2	0.01	0.23	Reagent.
argon	na	250	cu ft	g	1,2	1	250	Used to create an inert atmosphere.
boric acid	na	20	lb.	l,s	1,2	0.1	2	Reagent.
calcium carbonate anhydrous	na	22	lb.	s	1,2	0.01	0.22	Reagent.
calcium chloride	na	20	lb.	s	1	0.01	0.2	Reagent.
calcium nitrate	na	230	lb.	s	1	0.01	2.3	Reagent.
carbon	na	9	lb.	s	1,2	0.01	0.09	Reagent.
carbon dioxide	na	100	lb.	g	2	1	100	Reagent.
ceric ammonium nitrate	na	150	lb.	s	1,2	0.01	1.5	Reagent.
chem reagents, wet lab	--	<5	lb.	mixed	1			Assorted laboratory reagents in small quantities.
disodium phosphate	na	50	lb.	s	1,2	0.01	0.5	Reagent.
graphite	na	10	lb.	l	1,2	0.1	1	Reagent.
helium	na	213	cu ft	g	1,2	1	213	Used to create an inert atmosphere.
hydrofluoric acid gas	100	5	lb.	g	2	1	5	Reagent.
hydrogen fluoride	100	10	lb.	l	1	0.1	1	Reagent.
magnesium chloride	na	53	lb.	s	1,2	0.01	0.53	Reagent.
mercury	1	5	lb.	l	1,2	0.1	0.5	Use unknown.
mineral oil	na	77	lb.	l	1,2	0.1	7.7	Coolant and equipment lubricant.
nitric acid	1000	14	lb.	l	1,2	0.1	1.4	Reagent.
nitrogen	na	3270	lb.	g	2	1	3270	Used to create a nonflammable atmosphere.
oxalic acid	na	27	lb.	s	1,2	0.01	0.27	Reagent.
oxygen	na	220	cu ft	g	1,2	1	220	Welding.
paraffin	na	44	lb.	l	1,2	0.1	4.4	Shielding.
phosophorus pentoxide	na	7	lb.	s	1	0.01	0.07	Reagent.
phosphoric acid	5000	16	lb.	l	1,2	0.1	1.6	Reagent.
poly oedmi ethylene dichloride	1	4	gal	l	1,2	0.1	0.4	Use unknown, Dearcide 722 (14-730)
potassium chloride	na	110	lb.	s	1,2	0.01	1.1	Reagent.
potassium hydroxide	1000	64	gal	l	1,2	0.1	6.4	Use unknown, Dearborn 727 (1-688)
sodium borate	na	33	lb.	s	1,2	0.01	0.33	Reagent.
sodium carbonate	na	2107	lb.	s	1,2	0.01	21.07	Reagent.
sodium chloride	na	6	lb.	s	1,2	0.01	0.06	Reagent.
sodium hydroxide	1000	26	lb.	s	1,2	0.01	0.26	Reagent.
sodium hypochlorite	100	1	gal	l	1,2	0.1	0.1	Reagent.

Table A.18 (contd)

Facility/Chemical Name	RQ[a] lb.	Maximum Quantity Stored	Units[b]	Physical State[c]	Ref.[3]	Release Fraction[e]	Total Release	Comments
sodium nitrate	100	69	lb.	s	1,2	0.01	0.69	Reagent.
sodium sulfate	na	102	lb.	s	1,2	0.01	1.02	Reagent.
sulfamic acid	na	15	lb.	s	1,2	0.01	0.15	Reagent.
sulfur	na	100	lb.	s	1,2	0.01	1	Reagent.
sulfuric acid	1000	12	lb.	l	1,2	0.1	1.2	Reagent.
zinc	1	**2**	lb.	s	2	0.01	0.02	Reagent.
zinc nitrate	1000	100	lb.	s	1,2	0.01	1	Reagent.
zinc oxide	na	11	lb.	s	1,2	0.01	0.11	Reagent.
327 Bldg								
poly oedmi ethylene dichloride	1	**6**	gal	l	1	0.1	0.6	Use unknown, Dearcide 722 (14-222)
potassium hydroxide	1000	33	gal	l	1	0.1	3.3	Use unknown, Dearborn 727 (1-688)
trichloro-s-triazinetrione	na	50	lb.	l,s	1	0.1	5	Use unknown, Dearcide 730 (14-730)

a. RQ = CERCLA Reportable Quantity
b. lb. = pound; gal = gallon; cu ft = cubic feet
c. l = liquid; s = solid; g = gas
d. EPCRA reports 1992 (1); Bergsman 1995 (2); EPCRA tier II report, 1992 (3)
e. Fraction of stored chemical released in accidental spill scenario: 1.0 = gases; 0.1 = liquids; 0.01 = solids
f. NA = not applicable
bold = indicates a stored quantity that exceeds the RQ for that chemical
bold/italic = indicates an accidental release that exceeds the RQ for that chemical or chemical is highly toxic.

Table A-19. Baseline Chemical Inventory for Proposed Facilities

Facility/Chemical Name	RQ[a] (lb.)	Maximum Quantity Stored	Units[b]	Physical State[c]	Ref.[d]	Release Fraction [e]	Total Release	Comments
Wet Storage Facility								
argon	na[f]	42	lb.	g	b	1	42	Used to create an inert atmosphere.
chlorine	10	**9000**	lb.	s	b	0.01	***90***	For treatment of intake water.
EDTA disodium salt	na	267.5	lb.	s	b	0.01	2.675	Used for water analysis.
hydrogen peroxide	na	7	lb.	l	b	0.1	0.7	Cleaning and disinfection.
methane	na	42	lb.	g	b	1	42	Fuel.
nitrogen	na	6	lb	g	b	1	6	Used to create an inert atmosphere.
paraffin	na	1485	gal	l	b	0.1	148.5	Shielding and insulation.
PCB	1	**4701**	lb.	l	b	0.1	***470.1***	Transformer coolant.
potassium permanganate	100	8.8	lb.	s	b	0.01	0.088	Reagent.
sodium carbonate	na	2.2	lb.	s	b	0.01	0.022	Reagent and cleaner.
sodium hydroxide	1000	**3000**	gal	l	b,a	0.1	300	For water pH control.
sodium metabisulfite	na	4	lb.	s	b	0.01	0.04	Neutralizer.
stannous chloride	na	133	lb.	s	b	0.01	1.33	Reagent, catalyst, and cleaner.
sulfuric acid	1000	**3000**	gal	l	b,a	0.1	300	For water pH control.
Vault Dry Storage Facility								
argon	na	940	cu ft	g	a	1	940	Used to create an inert atmosphere.
decon soap	na	90	lb.	s,l	a	0.1	9	Decontamination of workers & equipment
Casks Dry Storage Facility								
decon soap	na	90	lb.	s,l	a	0.1	9	Decontamination of workers & equipment
Shear-Leach-Calcine Stabilization Facility								
argon	na	15200	lb.	g, l	a	1	15200	Used to create an inert atmosphere.
bromotrifluoromethane	na	1000	lb.	g	a	1	1000	Halon fire extinguishers.
ceramic formers	na	unk	lb.	s	a	0.01	unk	Solidifiers
diesel fuel	10	**20000**	lb.	l	a	0.1	***2000***	Fuel.
grease	na	100	lb.	l	a	0.1	10	Equipment lubricant
mineral oil	na	5000	lb.	l	a	0.1	500	Coolant and equipment lubricant.
nitric acid	1000	**1000000**	lb.	l	a	0.1	***100000***	Reagent.
nitrogen	na	1500	lb	l	a	1	1500	Used to create an inert atmosphere.
oxygen	na	100	lb.	g	a	1	100	Oxidizer
paraffin	na	200	lb.	l	a	0.1	20	Shielding and insulation.
propane	na	100	lb.	g	a	1	100	Fuel.
propylene glycol	na	200	lb.	l	a	0.1	20	Reagent.
sodium carbonate	na	1500	lb.	s	a	0.01	15	Reagent and cleaner.
sodium hydroxide	1000	**50000**	lb.	l	a	0.1	***5000***	For water pH control.
sodium nitrite	100	**5000**	lb.	l	a	0.1	***500***	Reagent.
sulfuric acid	1000	**25000**	lb.	l	a	0.1	***2500***	For water pH control.

Table A.19 (contd)

Facility/Chemical Name	RQ[a] (lb.)	Maximum Quantity Stored	Units[b]	Physical State[c]	Ref.[d]	Release Fraction [e]	Total Release	Comments
Solvent Extraction Fuel Stabilization Facility								
argon	na	15200	lb.	g, l	a	1	15200	Used to create an inert atmosphere.
bromotrifluoromethane	na	1000	lb.	g	a	1	1000	Halon fire extinguishers.
cadmium nitrate tetrahydrate	1	**1500**	lb.	l	a	0.1	***150***	Use unknown.
carbon dioxide	na	1000	lb.	g	a	1	1000	Use unknown.
diesel fuel	10	**20000**	lb.	l	a	0.1	***2000***	Fuel.
ferric nitrate	na	1000	lb.	s	a	0.01	10	Reagent.
ferris sulfamate	na	5000	lb.	l	a	0.1	500	Use unknown.
grease	na	100	lb.	l	a	0.1	10	Equipment lubricant
hydrazine	1	**1000**	lb.	l	a	0.1	***100***	Reagent.
hydrogen peroxide	na	1500	lb.	l	a	0.1	150	Cleaning and disinfection.
hydroxylamine nitrate	na	1500	lb.	l	a	0.1	150	Reagent.
kerosene	10	**40000**	lb.	l	a	0.1	***4000***	Fuel.
lubricating oil	na	678	lb.	l	a	0.1	67.8	equipment lubricant
mineral oil	na	5000	lb.	l	a	0.1	500	Coolant and equipment lubricant.
nitric acid	1000	**1000000**	lb.	l	a	0.1	***100000***	Reagent.
nitrogen	na	1500	lb	l	a	1	1500	Used to create an inert atmosphere.
oxalic acid	na	2000	lb.	s	a	0.01	20	Reagent.
oxygen	na	100	lb.	g	a	1	100	Oxidizer
paraffin	na	200	lb.	l	a	0.1	20	Shielding and insulation.
potassium permanganate	100	**1000**	lb.	s	a	0.01	10	Reagent.
propane	na	100	lb.	g	a	1	100	Fuel.
propylene glycol	na	200	lb.	l	a	0.1	20	Reagent.
sodium carbonate	na	1500	lb.	s	a	0.01	15	Reagent and cleaner.
sodium fluoride	1000	25	lb.	s	a	0.01	0.25	Use unknown.
sodium hydroxide	1000	**50000**	lb.	l	a	0.1	***5000***	For water pH control.
sodium nitrite	100	**5000**	lb.	l	a	0.1	***500***	Reagent.
sulfamic acid	na	5000	lb.	l	a	0.1	500	Reagent.
sulfuric acid	1000	**25000**	lb.	l	a	0.1	***2500***	For water pH control.
tartaric acid	na	2000	lb.	s	a	0.01	20	Reagent.
tributyl phosphate	na	5000	lb.	s	a	0.01	50	Reagent.
Fuel Drying and Passivation Facility								
argon	na	25200	lb.	g,l	a	1	25200	Used to create an inert atmosphere.
bromotrifluoromethane	na	1000	lb.	g	a	1	1000	Halon fire extinguishers.
carbon dioxide	na	1000	lb.	g	a	1	1000	Use unknown.
diesel fuel	10	**20000**	lb.	l	a	0.1	***2000***	Fuel.
grease	na	100	lb.	l	a	0.1	10	Equipment lubricant
mineral oil	na	5000	lb.	l	a	0.1	500	Coolant and equipment lubricant.
nitrogen	na	100000	lb	l	a	1	100000	Used to create an inert atmosphere.

Table A.19 (contd)

Facility/Chemical Name	RQ[a] (lb.)	Maximum Quantity Stored	Units[b]	Physical State[c]	Ref.[d]	Release Fraction [e]	Total Release	Comments
oxygen	na	10000	lb.	g	a	1	10000	Oxidizer
paraffin	na	200	lb.	l	a	0.1	20	Shielding and insulation.
propane	na	100	lb.	g	a	1	100	Fuel.
propylene glycol	na	200	lb.	l	a	0.1	20	Reagent.
sodium hydroxide	1000	**5000**	lb.	l	a	0.1	500	For water pH control.
sodium nitrite	100	**5000**	lb.	l	a	0.1	***500***	Reagent.
sulfuric acid	1000	**25000**	lb.	l	a	0.1	2500	For water pH control.

a. RQ = CERCLA Reportable Quantity
b. lb. = pound; gal = gallon; cu ft = cubic feet
c. l = liquid; s = solid; g = gas
d. a: Bergsman 1995 b: Chemical inventory of 105-KE Basin (as surrogate chemical inventory)
e. Fraction of stored chemical released in accidental spill scenario: 1.0 = gases; 0.1 = liquids; 0.01 = solids
f. NA = not applicable.

A.2.5 Toxicological Limits

Results from the EPIcode model were compared to available Emergency Response Planning Guideline (ERPG) values, Immediately Dangerous to Life and Health (IDLH) values, and Threshold Limit Values/Time-Weighted Averages. In the absence of these values, toxicological data for similar health endpoints, obtained from the Registry of Toxic Effects for Chemical Substances (RTEC), are used.

Emergency Response Planning Guidelines are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (DOE 1993b). Emergency Response Planning Guideline values are specific for a substance and are divided into three general severity levels: ERPG-1, ERPG-2, and ERPG-3. ERPG-1 values result in an unacceptable likelihood that one would experience mild transient adverse health effects or perception of a clearly defined objectionable odor (DOE 1993b). ERPG-2 values result in an unacceptable likelihood that one would experience or develop irreversible or other serious health effects or symptoms that could impair one's ability to take protective action (DOE 1993b). ERPG-3 values result in an unacceptable likelihood that one would experience life-threatening health effects (DOE 1993b).

For many chemicals, ERPG levels are not defined. In these instances, Threshold Limit Value/Time-Weighted Average (TLV/TWA) values are substituted for ERPG-1 values. Ten percent of Immediately Dangerous to Life or Health (IDLH) values are substituted for ERPG-2 values, and IDLH values are substituted for ERPG-3 values (DOE 1993b).

Data from RTEC were used for eight chemicals. Acute toxicity data were utilized to generate exposure limits to approximate the ERPG endpoints--irritation/odor, irreversible health effects, and death.

All references for Attachment A are included in Chapter 7 of this Appendix

ATTACHMENT B
EVALUATION OF OPTION FOR FOREIGN PROCESSING OF SPENT NUCLEAR FUEL CURRENTLY LOCATED AT THE HANFORD SITE

B.1 Description of Foreign Processing Alternative

This option was considered in response to a public comment requesting that foreign processing of N Reactor spent nuclear fuel (SNF) from the Hanford Site be addressed as a reasonable alternative to domestic stabilization and storage. Under this alternative, the SNF currently stored in basins at the 100-K Area of the Hanford Site would be packaged for shipment to an overseas facility where it would be processed. Only production reactor fuel stored at the 100-K Basins was considered in this analysis because it represents a large quantity of relatively homogenous material that would require stabilization in order to be suitable for 40-year storage. Small quantities of other types of fuel currently stored at Hanford either would not require stabilization or would have sufficiently different characteristics that they could not be stabilized efficiently by a single-process facility.

This analysis assumes that high-level waste (HLW) arising from the process would be returned to Hanford for interim storage, although it could potentially be stored overseas until a domestic repository was available in which to permanently dispose of it. Similarly, uranium and plutonium resulting from the processing were presumed to be returned to Hanford for interim storage; however, these materials could also be stored overseas until a decision is made on their disposition by the U.S. Department of Energy (DOE).

The following analysis was undertaken despite substantial uncertainties concerning the feasibility of long-distance transport of SNF in its current condition from the Hanford Site. Approximately half of the SNF is currently stored underwater at the 100-K West Basin in sealed, vented containers, and the remaining fuel is at 100-K East Basin in containers that are open to water. Efforts to characterize the physical and chemical state of the SNF are just getting underway, and those studies may reduce the uncertainties associated with long-distance transport of this SNF.

The SNF shipment would be required to meet national and international regulations specifying integrity of the cask seal in the event of internal pressure build-up, acceptable gas concentrations inside the cask, and allowable quantities of dispersible radionuclides. Because the defense production reactor SNF suffered damage during handling and discharge from the reactors, and because it was not designed for long-term durability in wet storage, a substantial fraction of the fuel elements have degraded during the time since reactor operations ceased (ranging from 7 to more than 20 years). The Hanford SNF in its present condition may not meet these requirements because of the quantity of dispersible radionuclides in damaged and corroding SNF, or because of heat generation and possible buildup of gases within the shipping container that might result from reactions between SNF and water in the wet overpack.

If the Hanford fuel were not able to meet the transportation requirements, the overseas processing alternative would necessitate additional expense and risk to stabilize the fuel or to divide the shipments into smaller quantities than assumed for the present analysis, perhaps to the extent that it might prove to be impractical altogether. The overland transport evaluation presented in Volume 1, Appendix I of this EIS assumed that Hanford SNF was in a stabilized form prior to shipment, as described in this appendix. Because of the uncertainties surrounding the feasibility of long-distance transport of Hanford SNF in its present condition, and to be consistent with the overland transport analysis in Appendix I, the SNF for overseas shipment is also presumed to be stabilized prior to shipment or is limited to elements that are sufficiently intact that the requirements of the transportation regulations could be met using a wet overpack shipping system. The shipment quantities assumed in the overseas transport analysis include the total mass of SNF estimated to be in the K Basins, although some of the SNF is known to exist as corrosion products and sludge, which would not be suitable for shipment without prior treatment to convert them into a less dispersible form.

B.2 Methods and Assumptions

The following sections describe the methods used to evaluate potential consequences of the overseas processing option. The analysis focuses on the activities associated with transportation of the SNF to the United Kingdom (U.K.) for processing and return of the waste and products to the U.S. The analysis also includes activities at Hanford to prepare the SNF for shipment, as well as those associated with transport and processing of the SNF within the U.K., to the extent that information was available. Information from an overseas processing facility located in the U.K. was used as the basis for this evaluation (BNFL 1994). However, the use of those facilities as a representative case would not preclude processing of SNF from Hanford at another suitable overseas installation.

B.2.1 Shipping Scenarios

Potential shipping scenarios are described in this option for transporting irradiated N Reactor fuel from the Hanford Site to the U.K., and the return of separated plutonium, uranium, and HLW to Hanford. All scenarios assume stabilization and packaging, as necessary, of the SNF currently stored in the 100-K Area Basins on the Hanford Site. From the 100 Area, the SNF would be loaded for onsite or offsite transport as required for each scenario. Offsite transport would take place via either barge, truck, or rail to a port designated as a "facility of particular hazard" in accordance with 33 CFR 126, where the shipment would be loaded onto a ship for overseas transport. The overseas segment of the shipment was assumed to utilize purpose-built ships typical of those employed by the representative processing facility in the U.K. for shipping SNF (BNFL 1994). Such a system would likely be necessary if Hanford SNF were to be shipped without prior stabilization because alternative carriers would presumably not have either the equipment or expertise required for long-distance transport of metallic SNF in a wet overpack. If the SNF were stabilized before shipment, a variety of commercial or military shipping options might be available (see DOE 1995 for a discussion of those options).

After processing of the SNF, the products and wastes were assumed to be returned to Hanford for interim storage via the same U.S. seaport at which the initial shipments exited the country. The three materials addressed in the analysis for the return shipments are plutonium, uranium, and HLW. It was assumed that the separated plutonium and uranium would be converted to oxide forms and shipped to the U.S. aboard a purpose-built ship similar to that used for transporting the irradiated fuel. Other transport options might also be available for these materials, including use of military or commercial ships or aircraft. High-level waste was assumed to be processed to a stable form (borosilicate glass encased in stainless steel canisters) before shipment. This section provides descriptions of the shipping scenarios, transportation and packaging systems, radiological characteristics of the shipments, transportation routes, and port facilities that were examined in this analysis.

B.2.1.1 Port Selection. Ports evaluated for the foreign processing option were chosen to minimize either the overland or ocean segments of the shipments and to provide a reasonable range of alternative transportation modes between the Hanford Site and the port (i.e., barge, truck, or rail). For the purposes of this evaluation, two potential West Coast U.S. ports (Seattle/Tacoma, Washington, and Portland, Oregon) and one potential East Coast port (Norfolk, Virginia) were evaluated for the overland transportation analysis. Population densities along the routes to these ports are representative of those in the vicinity of many major U.S. seaports. In addition, the port of Newark, New Jersey, was included in the port accident analysis to estimate the consequences of an accident in a location with a very high surrounding population.

B.2.1.2 Overseas Transport. The routing for overseas transport from West Coast U.S. ports would include transit via the Columbia River or Puget Sound to the Pacific Ocean, a southerly route through the Panama Canal or around Cape Horn in South America, and then north to the U.K. The route around the cape is considered because it maximizes the distance that a shipment might be required to travel, and therefore, provides an upper bound for risks associated with the ocean transport segment. However, a route via the Panama Canal would be preferable for West Coast shipments because it avoids potential risk associated with the added distance and adverse weather conditions that might be encountered during transport around the cape. Transport via an East Coast U.S. port would be directly across the Atlantic Ocean to the U.K. The total distance for ocean transport via the West Coast is approximately 7,000 nautical miles via the Panama Canal or 17,000 nautical miles via Cape Horn; that for the East Coast is approximately 3000 nautical miles.

B.2.1.3 Overland Transport Scenarios. Overland transport between the Hanford Site and overseas shipping ports was evaluated for three different scenarios, as described in the following sections.

B.2.1.3.1 Barge to Portland, Transoceanic Shipment to the U.K. This scenario begins with cask loading operations at the Hanford Site 100-K Area Basins. The shipping casks would be loaded with SNF and prepared for truck transport to the Port of Benton barge slip near the 300 Area of the Hanford Site. After arrival at the barge slip, the shipping casks would be transloaded onto the barge via crane and then secured to the deck of the barge. After a full load of casks was secured, the barge would depart for the Port of Portland, Oregon, traveling down the Columbia River through routinely navigated shipping channels. At the Port of Portland, the shipping casks would be lifted off the barge and placed aboard a ship for the overseas segment of the journey. The shipping casks would then be secured, and the ship would depart for the U.K. After processing of the SNF, the HLW shipments were assumed to return via Portland, where the material would be transloaded onto a rail car and transported to Hanford for interim storage. Shipments of uranium and plutonium oxide would be returned to Hanford by truck.

B.2.1.3.2 Truck/Rail to the Port of Seattle, Transoceanic Shipment to the U.K. The first leg of this scenario is different from the barge-to-Portland scenario in that the shipping casks would be loaded at the K Basins and shipped directly to the Port of Seattle, Washington, for trans-loading onto the ocean-going vessel. The overland leg would consist of either truck or rail shipments. It was assumed that one shipping cask would be transported per truck shipment or two casks per rail shipment. After arrival at the Port of Seattle, the shipping casks would be trans-loaded onto the ocean-going vessel and when a shipload of casks had been loaded, the ship would sail through Puget Sound and the Strait of Juan de Fuca to the Pacific Ocean, travel south via either the Panama Canal or Cape Horn, and then north to the U.K. After processing, the uranium, plutonium, and vitrified HLW would be returned to the U.S. by ship via Seattle and finally to Hanford by truck or rail.

B.2.1.3.3 Truck/Rail to the Port of Norfolk, Virginia, Transoceanic Shipment to the U.K. This scenario would be similar to the truck/rail to Seattle scenario except the intermediate port would be Norfolk, Virginia. Similar to the Port of Seattle scenario, the shipping casks would be loaded aboard the ocean-going vessel and shipped to the U.K. This shipping scenario maximizes the overland transport leg and minimizes the ocean travel distance. As with the other two shipping scenarios, the solidified HLW, plutonium oxide, and uranium oxide materials were assumed to be returned to Hanford via Norfolk.

B.2.2 Shipping System Descriptions

This section presents descriptions of the shipping cask and truck, rail, and barge shipping systems that are used in the three potential shipping scenarios. The information presented focuses on the parameters important to the impact calculations, namely the cargo capacities and radionuclide inventories.

The shipping cask assumed to be used for the SNF shipments from Hanford to the U.K. is a standard design routinely used for commercial SNF transport (BNFL 1994). The cask could transport approximately 5 tons of intact fuel (with a smaller capacity for damaged fuel). The loaded cask weight is about 46 tons, so it was assumed that one cask could be transported per highway shipment and two per rail shipment. The capacities of the barge and ship were assumed to be 24 casks each. A total of 17 transoceanic shipments would be required to accommodate the 408 caskloads that would be necessary to ship all Hanford SNF. The actual number of shipments required would depend on the number of casks available, or on procurement of a sufficient number of new casks to provide for efficient shipment of Hanford SNF on a reasonable schedule.

The radionuclide inventories for the SNF shipments were determined using the information on N Reactor fuel inventories presented in Bergsman (1994). The resulting radionuclide inventories for the three types of shipments (truck, rail, and barge/ship) are presented in Table B-1.

The return shipments of HLW and plutonium and uranium oxide were assumed to be shipped via the same routes used for overseas shipment of Hanford SNF. For the barge to Portland option, these materials were assumed to be returned to the U.S. by ship to the Port of Portland, where HLW shipping casks would be transloaded onto a barge and uranium and plutonium onto trucks for transport to Hanford. Similarly for the other options, the materials would be transported by ships to the ports of Norfolk or Seattle, transloaded onto truck or rail shipping systems, and transported to Hanford.

The number of shipments of solidified HLW was estimated using assumed shipping cask capacities for HLW. It is estimated that a total of 500 containers of vitrified HLW, each weighing about 500 kg, would result from processing the N Reactor SNF (BNFL 1994). The U.K. processing facility has designed a new 110-ton shipping cask for vitrified HLW that would be capable of carrying 21 HLW containers per shipment. Therefore, about 24 caskloads would be required to return the HLW to the U.S. This material was assumed to be transported to a U.S. port facility in one shipment and then transloaded onto a rail car for the overland shipment segment (the HLW cask is too large to be transported by regular truck service). The actual number of shipments required would depend on the number of HLW casks available or on procurement of a sufficient number of new casks to provide for efficient return shipment of HLW on a reasonable schedule.

The radionuclide inventories for the solidified HLW shipments are presented in Table B-1. These inventories were calculated by dividing the total quantity of each radionuclide shipped to the U.K. (exclusive of uranium and plutonium) by the number of HLW casks (24) to be returned to the U.S.

Table B-1. Facility and transport mode radionuclide inventory development[a]

Radionuclide	Curies/ MTU	Grams/ MTU	Total Curies in SNF	Curies/Shipment[b] Truck	Rail	Barge	Curies/Shipping Cask[c] HLW[d]	Plutonium Oxide[e]	Uranium Oxide[e]
Shipments				408	204	17	24/1	186	236
Duration				5 years	5 years	5 years	7 months	2.3 years	2.9 years
H3	4.59E+01		9.64E+04	2.36E+02	4.73E+02	5.67E+03	4.02E+03		
Fe-55	1.22E+01		2.56E+04	6.28E+01	1.26E+02	1.51E+03	1.07E+03		
Co-60	8.78E+00		1.84E+04	4.52E+01	9.04E+01	1.08E+03	7.68E+02		
Kr-85	8.07E+02		1.69E+06	4.15E+03	8.31E+03	9.97E+04	7.06E+04		
Sr-90	9.32E+03		1.96E+07	4.80E+04	9.59E+04	1.15E+06	8.16E+05		
Y-90	9.32E+03		1.96E+07	4.80E+04	9.59E+04	1.15E+06	8.16E+05		
Ru-106	8.52E+01		1.79E+05	4.39E+02	8.77E+02	1.05E+04	7.46E+03		
Rh-106	8.52E+01		1.79E+05	4.39E+02	8.77E+02	1.05E+04	7.46E+03		
Sb-125	2.02E+02		4.24E+05	1.04E+03	2.08E+03	2.50E+04	1.77E+04		
Te-125	4.94E+01		1.04E+05	2.54E+02	5.09E+02	6.10E+03	4.32E+03		
Cs-134	3.01E+02		6.32E+05	1.55E+03	3.10E+03	3.72E+04	2.63E+04		
Cs-137	1.20E+04		2.52E+07	6.18E+04	1.24E+05	1.48E+06	1.05E+06		
Ba-137m	1.14E+04		2.39E+07	5.87E+04	1.17E+05	1.41E+06	9.98E+05		
Ce-144	3.97E+01		8.34E+04	2.04E+02	4.09E+02	4.90E+03	3.47E+03		
Pr-144	3.97E+01		8.34E+04	2.04E+02	4.09E+02	4.90E+03	3.47E+03		
Pr-144m	4.77E-01		1.00E+03	2.46E+00	4.91E+00	5.89E+01	4.17E+01		
Pm-147	2.72E+03		5.71E+06	1.40E+04	2.80E+04	3.36E+05	2.38E+05		

Table B-1. (contd)

Radionuclide	Curies/ MTU	Grams/ MTU	Total Curies in SNF	Curies/Shipment[b] Truck	Curies/Shipment[b] Rail	Curies/Shipment[b] Barge	Curies/Shipping Cask[c] HLW[d]	Curies/Shipping Cask[c] Plutonium Oxide[e]	Curies/Shipping Cask[c] Uranium Oxide[e]
Shipments				408	204	17	24/1	186	236
Duration				5 years	5 years	5 years	7 months	2.3 years	2.9 years
Sm-151	1.10E+02		2.31E+05	5.66E+02	1.13E+03	1.36E+04	9.63E+03		
Eu-154	2.17E+02		4.56E+05	1.12E+03	2.23E+03	2.68E+04	1.90E+04		
Eu-155	5.14E+01		1.08E+05	2.65E+02	5.29E+02	6.35E+03	4.50E+03		
U-234	4.34E-01	6.94E+01	9.11E+02	2.23E+00	4.47E+00	5.36E+01			3.73E+00
U-235	1.60E-02	7.39E+03	3.35E+01	8.22E-02	1.64E-01	1.97E+00			1.37E-01
U-236	7.63E-02	1.18E+03	1.60E+02	3.93E-01	7.86E-01	9.43E+00			6.57E-01
U-238	3.31E-01	9.84E+05	6.94E+02	1.70E+00	3.40E+00	4.08E+01			2.85E+00
Np-237	4.75E-02		9.98E+01	2.45E-01	4.89E-01	5.87E+00	4.16E+00		
Pu-238	1.22E+02		2.56E+05	6.28E+02	1.26E+03	1.51E+04		1.33E+03	
Pu-239	1.36E+02	2.20E+03	2.86E+05	7.02E+02	1.40E+03	1.68E+04		1.48E+03	
Pu-240	9.94E+01	4.38E+02	2.09E+05	5.12E+02	1.02E+03	1.23E+04		1.08E+03	
Pu-241	8.71E+03	8.46E+01	1.83E+07	4.49E+04	8.97E+04	1.08E+06		9.48E+04	
Pu-242	6.45E-02	1.64E+01	1.35E+02	3.32E-01	6.63E-01	7.96E+00		7.01E-01	
Am-241	1.84E+02		3.86E+05	9.47E+02	1.89E+03	2.27E+04	1.61E+04		
Cm-244	2.62E+01		5.50E+04	1.35E+02	2.70E+02	3.24E+03	2.29E+03		

a. Radionuclide inventory taken from Bergsman (1994) and represents 10-year cooled Mark 1A fuel, in which Pu-240 constitutes 16% of total plutonium.

b. Curies/shipment inventories assume 1 cask per truck shipment, 2 truck casks per rail, and 24 truck casks per barge shipment.

c. Curies/cask inventories are based on one cask per truck and/or rail shipment.

d. HLW - Solidified high level waste; inventory assumes 100% removal of plutonium and uranium. High-level waste to be shipped only by barge (24 casks per barge) or rail (1 cask per rail car).

e. Plutonium and uranium oxide inventories assume 100% removal, and the number of shipments has been adjusted to reflect conversion from metal to oxide. Plutonium and uranium oxide to be shipped by barge and truck only.

The number of shipments of uranium and plutonium oxide were estimated using standard U.S. shipping equipment for uranium and plutonium. The estimated quantities to be shipped include 2,360 tons of purified uranium oxide and 6.5 tons of plutonium oxide generated from processing the K Basin SNF. For this analysis, it was assumed that the plutonium oxide would be transported by truck in a Type B package with a capacity of 35 kg/shipment. This results in a total of 186 caskloads of plutonium oxide. The vehicle for transport of plutonium was assumed to be a Safe-Secure Trailer/Armored Tractor specifically designed for shipment of special nuclear materials within the U.S. The uranium oxide was assumed to be transported by truck in shipping systems with a capacity of 10,000 kg/shipment. This would require a total of 236 caskloads of uranium oxide. One caskload per truck shipment for overland segments was assumed. One sea shipment of uranium oxide and one of plutonium oxide were assumed to be required.

The radionuclide inventories for the plutonium oxide and uranium oxide shipments are presented in Table B-1. The inventories were determined by dividing the total quantities of uranium and plutonium to be shipped to the U.K. by the respective numbers of caskloads presented above.

B.2.3 Transportation Route Information

The overland transportation routes assumed for this analysis are described in the following section. The descriptive information includes the shipping distances and population density data. These data were developed using the HIGHWAY (Johnson et al. 1993a) and INTERLINE (Johnson et al. 1993b) computer codes for truck and rail shipments, respectively, and are used to calculate transportation impacts. These data are summarized below for each transport segment described in Section B.2.2. No population data are presented for the ocean segments because once at sea, the exposed population becomes essentially zero.

Hanford to Seattle, Washington: The truck and rail shipping distances from Hanford to Seattle were determined to be 277 km (172 miles) and 716 km (445 miles), respectively. The large difference in shipping distance arises from the fact that the rail route is not a direct link to Seattle, but travels from Hanford to Vancouver, Washington and then to Seattle. For the highway route, the shipment travels through 88.1% rural areas (weighted population density 4.5 persons/km^2), 10% in suburban areas (359 persons/km^2) and 1.9% in urban population zones (1870 persons/km^2). The rail route travels through 74.1% rural areas (9.8 persons/km^2), 19% in suburban zones (415.5 persons/km^2), and 6.9% in urban areas (2226 persons/km^2).

Hanford to Norfolk, Virginia: The truck and rail shipping distances from Hanford to Norfolk were determined to be 4585 km (2849 miles) and 4984 km (3097 miles), respectively. For the highway route, the shipment travels through 84.5% rural areas (7.3 persons/km^2), 13.4% in suburban areas (365 persons/km^2) and 2.1% in urban population zones (2299 persons/km^2). The rail route travels through 83% rural areas (7.8 persons/km^2), 14.5% in suburban zones (360.4 persons/km^2), and 2.4% in urban areas (2149 persons/km^2).

Hanford to Portland, Oregon: The only option evaluated for using the Port of Portland was to barge the SNF to Portland, where it would be transloaded onto the ship. The distance and population density information for this shipment was approximated using INTERLINE (Johnson et al. 1993b), which evaluates potential rail routes, because the rail lines closely follow the Columbia River in which the barge would be operating. Consequently, the route data for a barge shipment would be similar to that for a rail shipment. The rail data are thought to be more conservative than actual barge data because the rail lines pass closer to the city centers along the river than would a barge.

B.2.4 Description of Methods Used to Estimate Consequences

This section describes the methods used to estimate consequences of normal and accidental exposure of individuals or populations to radioactive materials. The RADTRAN 4 (Neuhauser and Kanipe 1992) and RISKIND (Yuan et al. 1993) computer codes were used to calculate the transportation impacts, and the GENII software package (Napier et al. 1988) was used to estimate the consequences of port accidents. The MICROSHIELD external dosimetry software (Grove Engineering 1988) was used to determine approximate external dose rates for shipping containers as input to the transportation consequences. Nonradiological impacts from both incident-free transport and accidents were also evaluated.

The output from computer codes, as total effective dose equivalent (TEDE or dose) to the affected receptors, was then used to express the consequences in terms of potential latent cancer fatalities (LCF). Recommendations of the International Commission on Radiological Protection (ICRP 1991) for low dose, low dose rate radiological exposures were used to convert dose as TEDE to LCF. The conversion factor applied to adult workers was 4×10^{-4} LCF/rem TEDE, and that for the general population was 5×10^{-4} LCF/rem TEDE. The general population was assumed to have a higher rate of cancer induction for a given radiation dose than healthy adult workers because of the presence of more sensitive individuals (e.g., children) in the general population.

The estimated LCF for potential accidents was multiplied by the expected accident frequency per year, per shipment, or for the entire duration of the foreign processing operation, to provide a point estimate of risk consistent with those reported in the remainder of this EIS. Incident-free transportation or normal facility operations were assumed to occur (i.e., they have a frequency of 1.0); therefore, the cumulative risks associated with normal operations would be identical to the predicted number of latent cancer fatalities for the duration of the operation.

Nonradiological incident-free and accident impacts were also evaluated. Nonradiological incident-free impacts consist of fatalities from pollutants emitted from the vehicles. Nonradiological accident impacts are the fatalities resulting from potential vehicular accidents involving the shipments. Neither of these two categories of impacts are related to the radiological characteristics of the cargo. Estimates of these nonradiological impacts were derived by multiplying the unit risk factors (fatalities per mile of travel) by the total shipping distances for all of the shipments in each shipping option. Nonradiological unit risk factors for incident-free transport were taken from Rao et al. (1982), and for vehicular accidents were taken from Saricks and Kvitek (1994).

B.2.4.1 RADTRAN 4 Description. The RADTRAN 4 computer code (Neuhauser and Kanipe 1992) was used to perform the analyses of the radiological impacts of routine transport, the integrated population risks of accidents during transport of irradiated N-Reactor SNF to the U.K., and the return of vitrified HLW, plutonium oxide, and uranium oxide from the U.K. to Hanford. RADTRAN was developed by Sandia National Laboratories (SNL) to calculate the risks associated with the transportation of radioactive materials. The original code was written by SNL in 1977 in association with the preparation of NUREG-0170, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes* (NRC 1977). The code has since been refined and expanded and is currently maintained by SNL under contract with DOE. RADTRAN 4 is an update of the RADTRAN 3 (Madsen et al. 1986) and RADTRAN 2 (Taylor and Daniel 1982, Madsen et al. 1983) computer codes.

The RADTRAN 4 computer code is organized into the following seven models (Neuhauser and Kanipe 1992):

- material model

- transportation model
- population distribution model
- health effects model
- accident severity and package release model
- meteorological dispersion model
- economic model.

The code uses the first three models to calculate the potential population dose from normal, incident-free transportation and the first six models to calculate the risk to the population from user-defined accident scenarios. The economic model is not used in this study.

B.2.4.1.1 Material Model. The material model defines the source as either a point source or as a line source. For exposure distances less than twice the package dimension, the source is conservatively assumed to be a line source. For all other cases, the source is modeled as a point source that emits radiation equally in all directions.

The material model also contains a library of 59 isotopes each of which has 11 defining parameters that are used in the calculation of dose. The user can add isotopes not in the RADTRAN library by creating a data table in the input file consisting of eleven parameters.

B.2.4.1.2 Transportation Model. The transportation model allows the user to input descriptions of the transportation route. A transportation route may be divided into links or segments of the journey with information for each link on population density, mode of travel (e.g., trailer truck or ship), accident rate, vehicle speed, road type, vehicle density, and link length. Alternatively, the transportation route also can be described by aggregate route data for rural, urban, and suburban areas. For this analysis, the aggregate route method was used for each potential origin-destination combination. The origin-destination combinations addressed in this analysis were discussed in Section B.2.1.

B.2.4.1.3 Health Effects Model. The health effects model in RADTRAN 4 is outdated and is replaced by hand calculations. The health effects are determined by multiplying the population dose (person-rem) supplied by RADTRAN 4 by a conversion factor.

B.2.4.1.4 Accident Severity and Package Release Model. Accident analysis in RADTRAN 4 is performed using the accident severity and package release model. The user can define up to 20 severity categories for three population densities (urban, suburban, and rural), each increasing in magnitude. Eight severity categories for SNF containers that are related to fire, puncture, crush, and immersion environments are defined in NUREG-0170 (NRC 1977). Various other studies also have been performed for small packages (Clarke et al. 1976) and large packages (Dennis et al. 1978) that also can be used to generate severity categories. The accident scenarios are further defined by allowing the user to input release fractions and aerosol and respirable fractions for each severity category. These fractions are also a function of the physical-chemical properties of the materials being transported.

B.2.4.1.5 Meteorological Dispersion Model. RADTRAN 4 allows the user to choose two different methods for modeling the atmospheric transport of radionuclides after a potential accident. The user can input either Pasquill atmospheric-stability category data or averaged time-integrated concentrations. In this analysis, the dispersion of radionuclides after a potential accident is modeled by the use of time-integrated concentration values in downwind areas compiled from national averages by SNL.

B.2.4.1.6 Incident-Free Transport. The models described above are used by RADTRAN 4 to determine dose from incident-free transportation or risk from potential accidents. The public and worker doses calculated by RADTRAN 4 for incident-free transportation are dependent on the type of material being transported and the transportation index (TI) of the package or packages. The TI is defined in 49 CFR 173.403(bb) as the highest package dose rate in millirem per hour at a distance of 1 m from the external surface of the package. Dose consequences are also dependent on the size of the package, which as indicated in the material model description, will determine whether the package is modeled as a point source or line source for close-proximity exposures.

B.2.4.1.7 Analysis of Potential Accidents. The accident analysis performed in RADTRAN 4 calculates population doses for each accident severity category using six exposure pathway models. The exposure pathways are inhalation, resuspension, groundshine, cloudshine, ingestion, and direct exposure. This RADTRAN 4 analysis assumes that any contaminated area is either mitigated or public access controlled so the dose via the ingestion pathway equals zero. The consequences calculated for each severity category are multiplied by the appropriate frequencies for accidents in each category and summed to give a total point estimate of risk for a radiological accident. The parameters used to calculate the frequencies and consequences of transportation accidents are presented in Section B.2.4.2.

B.2.4.2 RADTRAN 4 Input Parameters. RADTRAN 4 input parameters for calculating routine population doses include route information (shipping distances, population densities, and fractions of travel in rural, suburban, and urban areas), numbers of shipments, dose rate, and parameters that define the population exposure characteristics. The route information and numbers of shipments were presented in Section B.1.2 and will not be repeated here. The remaining exposure parameters are described below.

RADTRAN 4 uses the dose rate at 1 m (referred to as the TI) in calculating dose to the public and worker. All of the SNF and HLW shipments in this analysis were assumed to be at the regulatory maximum dose rate, which is 10 mrem per hour at a distance of 2 m from the cask surface. This would be equivalent to a TI of 13 (or a dose rate of 13 mrem/hr at 1 m from the surface). Although it is likely that many of these shipments will have significantly smaller TI values, the use of the regulatory maximum value is bounding because it cannot be exceeded.

Because shipments of plutonium oxide and uranium oxide would have much smaller dose rates than SNF or HLW, preliminary shielding calculations were performed to derive more realistic values. The computer code MICROSHIELD (Grove Engineering 1988) was used to perform these calculations. Both types of shipments were modeled as cylindrical sources with cylindrical shields. The parameters used in these calculations are shown below:

- Plutonium oxide: The plutonium source was assumed to be 12.7 cm in diameter and 127 cm in length. Shielding was assumed to be provided by a 1-cm thick steel shield and an 8-cm thickness of solid hydrogenous material. The source inventory was the same as that shown in Table B-1.

- Uranium oxide: The uranium source was modeled as a single large container although the shipment will most likely be composed of several smaller containers. The source dimensions were assumed to be 114 cm in diameter and 370 cm in length. The source was assumed to be surrounded by a 1-cm thick steel cylinder and a 3-cm thick shield of solid hydrogenous material. The source inventory was shown in Table B-1.

The dose rate at 1 m from the surface of the plutonium oxide shipment was calculated to be 0.019 mrem/hr. Because this was increased by a factor of five to provide a bounding estimate, the TI value for these shipments was set to 0.1 mrem/hr. The dose rate for the uranium oxide shipments was calculated to be 0.0049 mrem/hr. This was also increased by a factor of five to 0.025 mrem/hr for conservatism.

Table B-2 is a list of input parameters that are used by RADTRAN 4 in the calculation of population dose for incident-free transportation. Many of the parameters are default values in the RADTRAN 4 code. Those that are not default values are identified and their sources are provided in footnotes to the table.

The potential receptors include workers and the general public. Worker doses include those received by the truck, rail, or barge crew and package handlers aboard the barge. Although RADTRAN models package handlers as persons who handle packages during intermediate stops, the routine doses to this group were assumed to apply to personnel who inspect the shipping containers aboard the barge. The equations used to calculate these doses assume that a five-person team spends approximately 0.5 hr per handling operation (or per inspection tour of the shipping casks). Although not exact, this is believed to be a reasonable approximation.

Table B-2. Input parameters for analysis of incident-free impacts[a]

Parameter	Rail	Barge	Truck
Dose rate 1 m from vehicle/package (mrem/h)[b]	13.1	13.1	13.1
Length of package (m)	3.0	3.0	3.0
Exclusive use	No	Yes	Yes
Velocity in rural population zone (km/h)[c]	64.4	16.09	88.6
Velocity in suburban population zone (km/h)[b]	40.3	8.06	40.3
Velocity in urban population zone (km/h)[c]	24.2	3.20	24.2
Number of crewmen	5	2	2
Distance from source to crew (m)	152	45.70	10.0
Stop time per km (h/km)[c]	0.033	0.01	0.011
Persons exposed while stopped[c]	100	50	50
Average exposure distance while stopped (m)[c]	20.0	50.0	20.0
Number of people per vehicle on link[c]	3	0	2
Traffic count passing a specific point-rural zone, one-way[c]	1.0	0	470
Traffic count passing a specific point-suburban zone, one-way[c]	5.0	0	780
Traffic count passing a specific point-urban zone, one-way[c]	5.0	0	2,800

a. Values shown are shipment-specific unless otherwise noted.
b. These values were used for SNF and HLW shipments. See text for the derivation of TI values for plutonium oxide (0.1 mrem/hr) and uranium oxide shipments (0.025 mrem/hr).
c. Default values from RADTRAN (Neuhauser and Kanipe 1992 and Madsen et al. 1983).

Public doses include doses to persons on the highway or railway (this category is not applicable to barge shipments as indicated in the RADTRAN documentation), doses to persons who reside near the highway, railway, or river, and doses at stops (for barge transport, this was assumed to include stops at navigation locks in dams). For all three shipping modes, the doses to passengers were assumed to be 0.0 because there would be no passengers traveling with the shipments. In addition, there were assumed to be no intermediate storage needs for the shipments, and the doses to in-transit storage personnel were set equal to 0.0.

Information needed to characterize the potential routes between Hanford and the U.K. include the shipping distances, population densities in rural, suburban, and urban areas along the routes, and fractions of total shipping distance that travel through rural, suburban, and urban areas. These data were presented in Section B.2.3.

B.2.4.3 RISKIND Description. RISKIND (Yuan et al. 1993) was used to calculate doses to the maximum individual and the public for both rail and truck transportation accidents. RISKIND was originally developed to model incident-free and accident conditions during transportation of SNF. The code was specifically designed to model accidental releases based on data contained in the NRC modal study (Fischer et al. 1987). RISKIND is designed to calculate the dose to individuals or groups of individuals for each of the severity categories identified in the modal study and provide probability-weighted dose risk, acute fatality, latent fatality, and genetic effect values. The probability-weighted dose risk values are calculated by multiplying and summing the dose for each severity category times the fraction of accidents within each severity category. Health effects are calculated by multiplying probability-weighted dose risk values by appropriate conversion factors. For this analysis, point estimates of risk for latent cancer fatalities were estimated as described in Section B.2.4.

The code is comprised of subroutines or models used to calculate radiological exposures to individuals at specific receptor locations. The information used to calculate these exposures can be performed using the default values contained in RISKIND or using receptor-specific data, supplied by the user. The exposure calculations are performed based on the receptor location, exposure conditions (i.e., inhalation and ingestion intake rates), and meteorological conditions.

RISKIND can be used to model all environmental exposure pathways based on the duration of the exposure. That is, for acute or short-term exposures, RISKIND can calculate exposures from initial plume passage or loss of shipping-cask shielding. For chronic or long-term exposures, RISKIND calculates exposures from ground deposition and ingestion from the food-chain pathways.

A radiological source inventory is contained internal to RISKIND that is based on fuel type, cooling times, and burnup rates. An analyst can input other radiological source inventories to calculate scenario-specific exposures. The radiological source inventory for this analysis is shown in Table B-1.

To calculate doses to the receptor, cask accident responses for both truck and rail, and release fractions have been incorporated into RISKIND. This information is based on the NRC modal study (Fischer et al. 1987). As discussed earlier, all shipments will be performed using Type B shipping containers; therefore, it is appropriate to use RISKIND to calculate the dose to the maximally exposed individual for all waste forms.

B.3 Radiological Dose to Workers

The following sections describe expected radiological consequences to workers during transportation and processing of N-Reactor SNF from Hanford.

B.3.1 Worker Dose from Pre-Shipment Activities at Hanford

Packaging of the K-Basin SNF for temporary wet storage was estimated to result in worker doses of approximately 140 person-rem (5.5×10^{-2} LCF) over a period of about 2 years. The activities covered by this estimate include repacking fuel assemblies in both K-East and K-West Basins and disposing of empty canisters (DOE 1992). The consequences of preparing the fuel for overseas shipment were assumed to be similar for the purposes of this evaluation. If stabilization of the fuel prior to shipment were necessary, an additional 180 person-rem might be accumulated by onsite workers over a 4-year period, resulting in 7.0×10^{-2} LCF (see Section 5.12.5 of this appendix). Consequences of air emissions from the storage or stabilization facilities to nearby workers would be much lower than those from direct exposure of workers in these facilities (see Section 5.7 of this appendix).

The consequences of accidents at the wet storage facility or the stabilization facility are discussed in Section 5.15 of this appendix. Air emissions from a fuel handling accident at the 100-K Basins or a uranium fire at the stabilization facility would result in a point estimate of risk to the nearby workers of $< 1.4 \times 10^{-7}$ LCF or $< 8.3 \times 10^{-12}$ LCF per year of operation, respectively. The estimated frequency for both accidents is between 1×10^{-6} and 1×10^{-4} per year. Operations at the K Basins to package SNF for shipment would last approximately 2 years, and the stabilization facility would require 4 years to process all of the K Basin SNF. The consequence to workers that might be directly involved in such accidents is highly speculative, and is addressed in Attachment A-Facility Accidents.

B.3.2 Worker Doses from Transportation to U.S. Ports

This section discusses the results of the worker impact calculations for truck, rail, and barge shipments to and from the U.K. These doses were calculated using the RADTRAN 4 computer code (Neuhauser and Kanipe 1992). The RADTRAN 4 program uses a combination of meteorological, demographic, health physics, transportation, packaging, and material factors to analyze risks associated with both normal transport (incident-free) and various user-selected accident scenarios. The RADTRAN 4 computer code description for both routine and accident impacts was presented in Section B.2.4.

The results of the incident-free transportation impact calculations are presented in Table B-3. The radiological impacts are presented in terms of the population dose (person-rem) received by exposed workers and the projected health effects calculated to occur in the exposed population. As shown, no excess fatalities were calculated to result from any of the five transportation options considered in this study.

As shown in Table B-3, the transportation option to U.S. ports that results in the lowest worker population doses is that involving barge shipments to the Port of Portland. This option is closely followed by the option of shipping by rail to the Port of Seattle. The option involving truck transport to the Port of Seattle is the third lowest option. The option of shipping by rail to the Port of Norfolk is next, followed by the option of shipping by truck to the Port of Norfolk. This result is intuitively obvious because the shipping distances are much longer from Hanford to Norfolk than to the other ports.

Table B-3. Results of incident-free transportation impact calculations for workers.

Option and material	Radiation doses, person-rem	Latent cancer fatalities
Barge to Portland		
SNF	3.0E+00	1.2E-03
HLW	1.8E-01	7.0E-05
Pu	7.7E-02	3.1E-05
U	5.3E-02	2.1E-05
TOTAL	3.3E+00	1.3E-03
Truck to Seattle		
SNF	6.0E+00	2.4E-03
HLW (Rail)	3.8E-01	1.5E-04
Pu (Truck)	4.5E-02	1.8E-05
U (Truck)	3.4E-02	1.3E-05
TOTAL	6.5E+00	2.6E-03
Rail to Seattle		
SNF	3.2E+00	1.3E-03
HLW (Rail)	3.8E-01	1.5E-04
Pu (Truck)	4.5E-02	1.8E-05
U (Truck)	3.4E-02	1.3E-05
TOTAL	3.7E+00	1.5E-03
Truck to Norfolk		
SNF	1.0E+02	4.2E-02
HLW (Rail)	1.5E+00	5.9E-04
Pu (Truck)	7.7E-01	3.1E-04
U (Truck)	5.8E-01	2.3E-04
TOTAL	1.1E+02	4.3E-02
Rail to Norfolk		
SNF	1.3E+01	5.0E-03
HLW (Rail)	1.5E+00	5.9E-04
Pu (Truck)	7.7E-01	3.1E-04
U (Truck)	5.8E-01	2.3E-04
TOTAL	1.5E+01	6.1E-03

In general, the shipments of N Reactor SNF to the U.K. would produce the highest doses of all the materials. This is attributed primarily to the higher number of N Reactor SNF shipments than the other materials. Also, it can be seen that rail shipments generally result in lower worker doses than truck shipments. This is because the exposure distances between the source and crew are much longer for rail shipments than for truck shipments. Similarly, the crew doses for rail and barge shipments are approximately comparable.

Maximum individual doses to workers from incident-free transport were calculated using the RISKIND computer code, consistent with the approach described in Volume 1, Appendix I. The maximally exposed workers for truck shipments were found to be the truck drivers (two-person crew), who were assumed to drive shipments for up to 2,000 hour per year. The maximally exposed worker for rail shipments was a transportation worker in a rail yard who spent a time- and distance-weighted average of 0.16 hours inspecting, classifying, and repairing railcars and was assumed to be present for all of the radioactive shipments.

The maximum incident-free exposure calculations for workers were performed for each shipping option. The results are 1.46 person-rem for the barge to Portland option, 2.0 person-rem for the option of shipping to Seattle by truck, 1.03 person-rem for the option of shipping to Seattle by rail, 35.3 person-rem for the option of shipping to Norfolk by truck, and 17.9 person-rem for the option of shipping to Norfolk by rail.

B.3.3 Worker Dose from Port Activities

The following sections describe expected radiological consequences to workers from in-port activities for transport of SNF to the U.K. The consequences for return of HLW, uranium, and plutonium are expected to be similar to, or lower than, those for initial shipment of SNF to the U.K. because of the smaller number of HLW shipments required for return to the U.S. Radiological consequences of normal transport of uranium and plutonium would be small compared with those for SNF and HLW.

B.3.3.1 Consequences of Normal Port Activities. Consequences to workers during handling and loading activities in ports are based on commercial experience during the last three quarters of 1994. Over this period, workers handled two shipments consisting of 16 loaded casks, and 1 shipment consisting of 5 empty casks. The collective dose to the 30 workers involved was 0.024 person-rem, with the maximum individual receiving 0.016 rem. Assuming that handling of the empty casks did not contribute measurably to that total, the expected collective dose from handling a single loaded cask is estimated to be on the order of 0.001 rem to the maximally exposed worker and 0.0015 person-rem total to all workers. The consequences for loading and unloading of 408 casks during shipment from the U.S. to the U.K. would therefore be approximately 1.2 person-rem to all workers over the expected 5-year campaign. Accounting for an additional two handling activities per cask at the Hanford Site and at the U.K. process facility would roughly double that estimate, resulting in a collective dose of 2.4 person-rem and a potential for 9.8×10^{-4} LCF for all shipments. The maximum dose to an individual worker, assuming that worker were involved in handling all 408 casks at one point in the shipping sequence, would be on the order of 0.4 rem over 5 years.

B.3.3.2 Consequences of Accidents During Port Activities. The consequences of accidents during port transit were estimated based on the highest activity N Reactor SNF (Bergsman 1994). The assumed radionuclide content of a single shipping cask is based on a loading of 5 MTU (see inventory for truck shipments in Table B-1). Representative ports on the West and East Coasts of the U.S. (Seattle-Tacoma, Washington; Portland, Oregon; Norfolk, Virginia; and Newark, New Jersey) were used for this analysis, based on relative population densities and suitability for handling of SNF shipments. Newark was included in this part of the analysis because of its relatively large surrounding population (adjacent to New York City), whereas the ports of Seattle-Tacoma, Portland, and Norfolk are located in somewhat smaller population centers. In a previous analysis, the collective consequences of in-port accidents were shown to be proportional to the surrounding population (DOE 1995).

The consequences (as radiation dose to individuals and populations and corresponding LCF were evaluated for a range of accident severities leading to airborne release of radioactive material, corresponding to the accident categories and radionuclide release fractions used for the overland transportation analysis (Volume 1, Appendix I, Table I-28). The overall accident frequency associated with each accident category was calculated using the conditional probability for that severity category, multiplied by the overall frequency with which a shipping accident would occur (as estimated by DOE 1994, Table E-8). The consequences (as LCF) for each severity category were multiplied by the corresponding frequency with which an accident in that category would occur to obtain a point estimate of risk for each accident category. The total risk per shipment was then calculated as the sum of risks over all accident severity categories. The frequencies for airborne release accidents evaluated using 95% atmospheric dispersion (stable) conditions (those that would not be exceeded more than 5% of the time) were assumed to be 10% of those evaluated using 50% (neutral) dispersion conditions, which are assumed to be the typical or expected conditions. The risk to U.S. ports for shipping all Hanford SNF overseas is the total risk per shipment times 17 shipments. The risk to U.K. ports is assumed to be comparable to that at U.S. ports.

The port accident analyses assume that the contents of a single cask were involved in any given accident. The probability that multiple casks could be breached in the event of an accident is smaller than that for a single cask, and the consequences would be proportional to the number of casks involved. Because of the construction of the special purpose ships, with eight segregated holds each containing at most three casks, an accident that would involve more than three casks is not considered to be reasonably foreseeable.

The consequences to an individual at a distance of 100 m, assumed to be a port worker, was estimated for applicable exposure pathways including inhalation, external dose from submersion in the plume, and external exposure from radionuclides deposited on the ground for a period of 2 hours. The point estimates of risk for an accident at the Port of Portland are estimated to be 6.1×10^{-11} to 1.0×10^{-09} LCF for 1 to 17 shipments, respectively. The corresponding point estimates of risk for Seattle/Tacoma (based on wind data from Seattle-Tacoma airport and the population within 50 miles of the Port of Tacoma) ranged from 4.7×10^{-11} to 8.0×10^{-10} LCF. The point estimates of risk to workers at East Coast ports were similar - ranging from 6.1×10^{-11} to 1.0×10^{-09} LCF at Norfolk and 5.3×10^{-11} to 9.0×10^{-10} LCF at Newark.

The maximum reasonably foreseeable accident was a category 6 accident, which has a frequency of 1.3×10^{-7} per port transit, and which was evaluated for stable atmospheric conditions resulting in a cumulative frequency of 2.2×10^{-7} for all 17 SNF shipments. The dose to the port worker was estimated to be 1.7 rem at Seattle/Tacoma, 1.9 rem at Newark, and 2.1 rem at Portland and Norfolk. The corresponding probability of LCF ranged from 6.8×10^{-4} and point estimates of risk, from 1.5×10^{-9} to 1.8×10^{-9} LCF.

B.3.4 Worker Dose from Ocean Transport to the United Kingdom

The following sections describe radiological consequences to workers from normal transport operations and accidents during overseas shipments of SNF from the Hanford Site to the U.K.

B.3.4.1 Consequences of Normal Ocean Transit. The primary impact of routine (incident-free) marine transport of SNF is potential radiological exposure to crew members of the ships used to carry the casks. Members of the general public and marine life would not receive any measurable dose from the SNF during incident-free marine transport of the casks. While at sea, the crew dose would be limited to those individuals who might enter the ship's hold during transit and receive external radiation in the vicinity of the packaged SNF. At all other times, the crew would be shielded from the casks by the decking and other structures of the vessel. The number of entries and inspections would be a function of the transit time from the port of loading to the port of off-loading.

External radiation from an intact shipping package must be less than specified limits that control the exposure of the handling personnel and general public. These limits are established in 49 CFR Part 173. The limit of interest is a 10 mrem/hr dose rate at any point 2 m from the outer surfaces of the transport cask. This limit applies to exclusive-use shipments, i.e., a shipment in which no other cargo is loaded on the platform used for the transportation casks, not that the ship is an exclusive-use vessel, although this would not be a limitation for the commercial special purpose ships assumed for this analysis.

It is anticipated that the external dose rates at the outside of the transport casks would be much less than the regulatory limits. It was estimated that the N Reactor SNF considered in this analysis would fall within the design envelope of the internationally licensed casks routinely used by the U.K. facility for SNF transport (BNFL 1994). However, estimates of dose during normal transportation have been made assuming dose rates at the regulatory limits, using analyses performed for transport of foreign research reactor SNF as a basis (DOE 1995). These analyses may be used to develop an upper bound of the doses anticipated to be received by ships crews during transport of the N Reactor SNF. Actual doses would be expected to be lower than these estimates.

B.3.4.1.1 Bounding Dose Calculations. Calculations performed to estimate bounding radiation doses during routine cask inspections aboard ship (DOE 1995) provided information from which an inspection dose factor (IDF) could be determined of 6×10^{-5} rem $\cdot$ minute^{-1} $\cdot$ cask^{-1} $\cdot$ day^{-1} $\cdot$ person^{-1}, based on an average distance of 5.5 m. Because the ship crews are highly trained and the ships are designed for SNF transport, it was assumed that inspection of each of the eight holds on the ship (each containing three casks) would take no longer than 15 minutes, or an average of 5 minutes per cask for the total 24 casks. The total inspection time per day would be 2 hours. If an inspection crew were assumed to consist of two members of the ship's crew, the bounding dose per daily inspection would be

$$6 \times 10^{-5} \text{ (IDF)} \times 5 \text{ minutes} \times 24 \text{ casks} = 0.007 \text{ rem} \cdot \text{person}^{-1} \cdot \text{day}^{-1} \qquad (1)$$

Assuming a travel time from an eastern U.S. port of 10 days, the estimated maximum dose received by each member of a two-person inspection crew would be 0.07 rem. This value would not exceed the 0.1 rem dose limit for a member of the general public. The transit time for a shipment originating on the West Coast of the U.S. could be up to five times longer, resulting in a dose per shipment of 0.35 rem. This value would exceed the 0.1 rem dose limit for a member of the general public. However, because the ship's crews are trained and issued dosimeters, it is presumed that they would be considered radiation workers. Although it is not clear at this time if radiation exposure of the ship's crew would fall under the jurisdiction of the U.K. or U.S. radiation protection standards, these standards are identical for both countries (5 rem per year, with an administrative control level of 2 rem per year). Therefore, the maximum possible dose received by individual workers during ocean transit would be well within the limits of the U.S. and U.K. radiation protection standards for workers.

Complete transport of the SNF to the U.K. for processing would require 17 shipments of 24 casks. The collective dose to crew members responsible for conducting inspections on the transport ships during fuel transport from the U.S. East Coast would be

$$(0.007 \text{ rem} \cdot \text{person}^{-1} \cdot \text{day}^{-1}) \times 2 \text{ persons} \times (10 \text{ days} \cdot \text{trip}^{-1}) \times 17 \text{ trips} = 2.4 \text{ person-rem} \qquad (2)$$

Based on this bounding estimate of the collective dose to the ship's crew for transportation of the SNF, an upper limit of approximately 0.001 LCF would be expected among the ship's crew from exposure to external radiation from the SNF transport casks. If all shipments originated at a western U.S. port, the collective dose could be up to 12 person-rem with a corresponding consequence of 0.005 LCF.

The above analysis does not consider the return of the processed SNF products and waste from the U.K. to the U.S. It was projected that the number of shipments containing these products would be fewer than the number of SNF shipments. However, as a bounding estimate the same number of return shipments and similar external dose rates, at the regulatory limit, might be assumed. Under those circumstances, an upper limit of 0.01 LCF would be expected among the ships' crews from exposure to the external radiation during all shipments.

B.3.4.1.2 Commercial Fuel Transport Experience. Information on radiation doses to ships' crews during transport of commercial fuel, gathered from actual crew dosimeters, supports the statements above that actual doses to the crew would be lower than the calculated bounding doses. The average individual dose during one voyage was 0.001 rem, with a maximum individual dose of 0.022 mrem. The collective dose to the ship's crew for one voyage was about 0.038 person-rem. On that basis, the crew's collective dose for 17 SNF shipments would be 0.65 person-rem. A comparison of bounding dose estimates and commercial transport experience is shown in Table B-4. Based on these results, less than 0.0003 LCF would be expected among ships' crews

Table B-4. Comparison of bounding and typical ship crew's doses.

	Bounding Dose Calculations	Commercial Fuel Transport Experience
Individual dose, rem	0.07 - 0.35	0.001 typical 0.022 maximum
Collective dose, person-rem		
- 17 SNF shipments	2.4 - 12	0.65
- $\leq$ 17 round trips	$\leq$ 24	$\leq$ 1.3

from radiation exposure during SNF transport, and approximately 0.0005 LCF would be expected from radiation exposure during transport of SNF and the subsequent return of processing products and waste.

B.3.4.2 Consequences of Accidents During Ocean Transit. The consequences of accidents during ocean transit would likely be similar to those of port workers who are near the scene of an accident (see Section B.3.3.2). Individuals in the immediate vicinity of the impact would probably not survive an accident severe enough to cause release of radioactive materials from a SNF shipping cask. Effects on the ocean environment would not be expected to be discernable because of the degree of dispersion in the event of an airborne release.

B.3.5 Worker Dose from Return of Processing Products to the United States

Return of HLW to the U.S. is assumed to result in cumulative worker doses that are bounded by those incurred in the initial SNF shipments to the U.K. However, the distribution of dose among individual workers may differ because of the different configuration and radionuclide content of the HLW canisters. As noted in Section B.2.4.2, the dose rates associated with plutonium and uranium shipments are substantially below the regulatory maximum that was assumed for the SNF and HLW shipments.

B.4 Consequences to Members of the Public

The following sections describe expected consequences to the public from various activities involved in transporting N Reactor SNF to the U.K.

B.4.1 Public Impacts from Pre-Shipment Activities at Hanford

Activities at Hanford prior to preparation of N Reactor SNF for shipment would result in generally small consequences to the public, as discussed in Section 5.7 of this appendix. The removal and packaging of SNF at the basins was estimated to result in offsite consequences comparable to those observed during initial segregation of the fuel, or approximately 2×10^{-5} to 3×10^{-4} (1×10^{-11} to 1.5×10^{-10} probability of LCF) mrem to the maximally exposed offsite individual (DOE 1992).

The risk from accidents involving handling of N-Reactor SNF at the 100-K Basins was also presented in Section 5.15 of this appendix. The consequences to the maximally exposed offsite individual were estimated as 2.5×10^{-4} LCF, with an associated point estimate of risk equal to $<2.5 \times 10^{-8}$ fatal cancers per year (assuming an accident frequency $< 1 \times 10^{-4}$ per year). The consequences to the population within 80 km (50 miles) were estimated as 0.4 LCF for 50% (neutral) atmospheric dispersion conditions and 6.9 LCF for 95% (stable) atmospheric dispersion (conditions that would not be exceeded more than 50% or 5% of the time, respectively). The corresponding point estimates of risk amounted to $<4.0 \times 10^{-5}$ and $<6.9 \times 10^{-4}$ LCF per year, respectively.

B.4.2 Public Impacts from Transportation Activities

This section presents the analysis of the public incident-free radiological exposures, radiological accident risks, and nonradiological impacts from transporting radioactive materials to and from the U.K. Members of the public exposed to radiation include persons on the highway, railroad, or waterway with the shipment, persons residing near these transport links, and persons at intermediate stops along the route (such as refueling stops and stops at rail classification yards). The RADTRAN 4 computer code was used to perform these calculations. A description of RADTRAN 4 was presented in Section B.2.4. The following sections present the results of the incident-free exposure calculations, description of the accident-analysis input parameters, the results of the accident risk impact calculations, and the evaluation of nonradiological impacts.

B.4.2.1 Results of Incident-Free Transportation Impact Calculations. The results of the public dose calculations, developed using the RADTRAN 4 computer code and the input parameters described in Section B.2.4, are presented in Table B-5.

Table B-5. Results of public incident-free exposure calculations.

Option and material	Radiation doses, person-rem	Latent Cancer Fatalities
Barge to Portland		
SNF	3.4E-01	1.7E-04
HLW	6.7E-03	3.4E-06
Pu	3.7E-02	1.9E-05
U	2.9E-02	1.4E-05
TOTAL	4.1E-01	2.1E-04
Truck to Seattle		
SNF	1.5E+01	7.6E-03
HLW (rail)	1.9E-01	9.6E-05
Pu (truck)	2.5E-02	1.2E-05
U (truck)	1.9E-02	9.3E-06
TOTAL	1.5E+01	7.7E-03
Rail to Seattle		
SNF	1.6E+00	8.1E-04
HLW (rail)	1.9E-01	9.6E-05
Pu (truck)	2.5E-02	1.2E-05
U (truck)	1.9E-02	9.3E-06
TOTAL	1.9E+00	9.3E-04
Truck to Norfolk		
SNF	2.5E+02	1.3E-01
HLW (rail)	7.0E-01	3.5E-04
Pu (truck)	4.1E-01	2.1E-04
U (truck)	3.1E-01	1.6E-04
TOTAL	2.5E+02	1.3E-01
Rail to Norfolk		
SNF	5.9E+00	3.0E-03
HLW (rail)	7.0E-01	3.5E-04
Pu (truck)	4.1E-01	2.1E-04
U (truck)	3.1E-01	1.6E-04
TOTAL	7.3E+00	3.7E-03

From a domestic transportation perspective, the lowest-impact option is one that includes rail shipments of SNF from Hanford to the Port of Seattle. This option is followed closely by the option of moving SNF from Hanford to the Port of Portland by barge. The third lowest domestic transportation option is that involving SNF shipments to Seattle by truck. The highest impact options are those involving shipments from Hanford to the Port of Norfolk. Obviously, the lowest impact domestic transportation option would be that involving the shortest shipping distances (i.e., Hanford to Seattle or Portland). Some of the impacts of the long domestic transportation links would be offset by subsequent reductions in the lengths of the ocean shipment segments. Consequently, the rankings of the options presented in Table B-5 do not necessarily represent the rankings that would result if the ocean segments of the shipments were included. However, public routine doses are not significant for ocean voyages because the separation distance between the ship and the nearest exposed population is greater, resulting in extremely low radiation dose rates.

The results in Table B-5 demonstrate that barge shipments of SNF (and HLW) would produce lower public routine doses than truck or rail shipments. This is attributed primarily to the lower traffic volumes on waterways relative to railroads and highways, generally greater separation distances between barges and the public relative to the separation distances between highways/ railroads and the public, as well as the increased per-shipment capacities of barges relative to truck and rail shipments (resulting in fewer shipments).

Table B-5 also demonstrates that rail shipments would produce lower public routine doses than equivalent truck shipments. This can be seen by comparing the SNF shipment impacts for truck shipments to Seattle (15 person-rem) and rail shipments to Seattle (1.6 person-rem). Even though the rail shipping route from Hanford to Seattle is much longer than the truck route (277 km and 716 km), the total public routine doses are smaller. As with barge shipments, this is attributed to lower traffic volumes, larger separation distances, and increased shipment capacity for rail shipments.

Maximum individual doses to members of the public from incident-free transport were calculated using the RISKIND computer code, which is consistent with the approach described in Volume 1, Appendix I. For rail shipments, three potential exposure scenarios were evaluated by RISKIND, as described in Volume 1, Appendix I. The maximally exposed members of the public from incident-free truck transport were also determined using three potential exposure scenarios (see Volume 1, Appendix I).

The maximum incident-free exposure calculations for members of the public were performed for each shipping option. The results are 0.28 person-rem for the barge to Portland option, 0.20 person-rem for the option of shipping to Seattle by truck, 0.28 person-rem for the option of shipping to Seattle by rail, 0.20 person-rem for the option of shipping to Norfolk by truck, and 0.28 person-rem for the option of shipping to Norfolk by rail.

B.4.2.2 Assessment of Public Impacts from Transportation Accidents. Radiological accident impacts are presented in this section as integrated population risks (i.e., accident frequencies multiplied by consequences integrated over the entire shipping campaign), as well as the consequences of the maximum reasonably foreseeable accident. Population risk calculations were performed using the RADTRAN 4 computer code (Neuhauser and Kanipe 1992). The consequences of the maximum reasonably foreseeable accident were calculated using the RISKIND computer code (Yuan et al. 1993). Separate sections are provided for the integrated population risk (i.e., RADTRAN 4) calculations and the maximum reasonably foreseeable accident consequence (i.e., RISKIND) calculations.

B.4.2.2.1 Integrated Population Risk Assessment. For this analysis, risk is defined as the product of the frequency of occurrence of an accident involving a shipment and the consequences of an accident. Consequences are expressed in terms of the radiological dose and LCF from a release of radioactive material from the shipping cask or the exposure of persons to radiation that could result from damaged package shielding. The frequency of an accident that involves radioactive materials is expressed in terms of the expected number of accidents per unit distance integrated over the total distance traveled. The response of the shipping cask to the accident environment and the probability of release or loss of shielding, is related to the severity of the accident.

The frequencies of occurrence of transportation accidents that would release significant quantities of radioactive material are relatively small because the shipping casks are designed to withstand specified transportation accident conditions (i.e., the shipping casks for all the materials shipped in this analysis were assumed to meet the Type B packaging requirements specified in 49 CFR 174 and 10 CFR 71). Accidents on the road and railways are difficult to totally eliminate. However, because the shipping casks are capable of withstanding certain accident environments, including mechanical and thermal stress, only a relatively small fraction of accidents involve conditions that are severe enough to result in a release of radioactive materials.

Should an accident involving a shipment occur, a release of radioactive material could occur only if the cask were to fail. A failure would most likely be a small gap in a seal or small split in the containment vessel. For the radioactive material to reach the environment, it would have to pass through the split in the cask or through the failed seal. Materials released to the environment would be dispersed and diluted by weather action and a fraction would be deposited on the ground (i.e., drop out of the contaminated plume) in the surrounding region. Emergency response crews arriving on the scene would evacuate and secure the area to exclude bystanders from the accident scene. The released material would then be cleaned up using standard decontamination techniques, such as excavation and removal of contaminated soil. Monitoring of the area would be performed to locate contaminated areas and to guide cleanup crews in their choice of protective clothing and equipment (e.g., fresh-air equipment and filtered masks). Access to the area would be restricted by federal and/or state radiation control agencies until it had been decontaminated to safe levels.

The RADTRAN 4 computer code was used to calculate the radiological risk of transportation accidents involving radioactive material shipments. The RADTRAN 4 methodology was summarized previously. For further details, refer to the discussions presented by *RADTRAN III* (Madsen et al. 1986) and *RADTRAN 4: Volume 2 – Technical Manual* (Neuhauser and Kanipe 1992).

There are five major categories of input data needed to calculate potential accident transportation risk impacts using the RADTRAN 4 computer code. These are: 1) accident frequency, 2) release quantities, 3) atmospheric dispersion parameters, 4) population distribution parameters, and 5) human uptake and dosimetry models. Accident frequency and release quantities are discussed below, the remaining parameters have been discussed in previous sections.

Accident Frequency. The frequency of a severe accident is calculated by multiplying an overall accident rate (accidents per truck-km or per rail-km) by the conditional probability that an accident would involve mechanical and/or thermal conditions that are severe enough to result in container failure and subsequent release of radioactive material. Overall accident rates per kilometer of truck or rail travel were taken from Saricks and Kvitek (1994). State-specific accident rates were used in this study. For the Portland and Norfolk options, a composite weighted-average accident rate was developed using the state-specific accident rates in Saricks and Kvitek (1994), and travel fractions through each state that were derived from the HIGHWAY and INTERLINE results.

For this analysis, six shipment-specific severity categories were defined, with category 1 as the least severe and the higher categories (2-6) representing increasingly severe conditions. The conditional probabilities of encountering accident conditions in each severity category were taken from a U.S. Nuclear Regulatory Commission (NRC) document (Fischer et al. 1987). Those conditional probabilities were developed based on reviews of accident records and statistics compiled by various state and federal agencies. The conditional probability for a given severity category is defined as the fraction of accidents that would fall into that severity category if an accident were to occur. The conditional probabilities for truck and rail shipments were determined using a binning process described in Volume 1, Appendix I of this EIS. The derivation of the accident rates and conditional probabilities used in this analysis are discussed below. [The conditional probabilities for barge accidents were taken directly from Pippen et al. (1995)].

As discussed above, severity category levels were defined to model the response of the various shipments to accidents. Severity category 1 was defined as encompassing all accidents that are within the type B package envelope that would not be severe enough to result in failure of the shipping cask (i.e., accidents with zero release). The higher categories (2-6) were defined to include more severe accidents, and thus may lead to a release of radioactive material. The derivation of the severity category schemes and conditional probabilities of accidents in each severity category are discussed below for each shipping cask or container type. Table B-6 presents the conditional probabilities of the various severity categories that were used in this analysis.

Release Fractions. Release fractions (array RFRAC in RADTRAN 4) are used to determine the quantity of radioactive material released to the environment as a result of an accident. The quantity of material released is a function of the severity of the accident (i.e., thermal and mechanical conditions produced in the accident), the response of the shipping container to these conditions, and the physical and chemical properties of the material being shipped. The basis for the release fractions used in this analysis are discussed below and summarized in Table B-7.

Release fractions for N Reactor fuel shipments were taken from Volume 1, Appendix I of this EIS. The table of release fractions for metallic fuels was used (Table I-28). All of the released material was assumed to be in respirable form for this assessment. Release fractions for damaged N Reactor SNF were modeled the same as for undamaged fuel. This is because it was assumed that some form of stabilization would occur prior to shipment of damaged SNF. Stabilization was assumed to provide a level of containment for damaged SNF, such as placement in an overpack container, to replace the containment boundary that was provided by the failed N Reactor SNF cladding. Stabilization was also assumed to include some form of treatment to minimize the likelihood of a pyrophoric reaction involving the metallic uranium and to prevent the accumulation of an explosive concentration of hydrogen gas that may be generated by the fuel elements.

Table B-6. Accident severity categories and conditional probabilities.

Mode	Conditional probability by severity category					
	1	2	3	4	5	6
Truck[a]	9.943E-01	4.03E-05	3.82E-03	1.55E-05	1.80E-03	9.84E-06
Rail[a]	9.940E-01	2.02E-03	2.72E-03	6.14E-04	8.55E-04	1.25E-04
Barge[b]	9.53E-01	2.02E-03	4.02E-02	6.41E-04	4.01E-03	1.34E-04
Ship[c]	6.03E-01	3.95E-01	2.0E-03	4.0E-04	4.0E-04	4.0E-04

a. Source: Fischer et al. (1987) and Volume 1, Appendix I, Figure I-2.
b. Source: Pippen et al. (1995).
c. Source: DOE (1994).

Table B-7. Release fractions used for assessment of accident impacts.

Material	Release fraction by severity category					
	1	2	3	4	5	6
SNF[a]						
Gases	0.0	9.9E-03	3.3E-02	3.9E-01	3.3E-01	6.3E-01
Cesium	0.0	3.0E-08	1.0E-07	1.0E-06	1.0E-06	1.0E-05
Ruthenium	0.0	4.1E-09	1.4E-08	2.4E-07	1.4E-07	2.4E-06
Particles	0.0	3.0E-10	1.0E-09	1.0E-08	1.0E-08	1.0E-07
HLW[a]	HLW release fractions are the same as those for SNF					
Pu oxide						
Particles	0.0	1.0E-06	1.0E-05	1.0E-04	1.0E-03	1.0E-02
U oxide						
Particles	0.0	1.0E-06	1.0E-05	1.0E-04	1.0E-03	1.0E-02

a. These release fractions were applied to truck and rail shipments of SNF and HLW. Release fractions for barge shipments were multiplied by 1/24, 1/12, 1/6, 1/3, and 1 for severity categories 2 through 6, respectively, to reflect the number of shipping casks that are damaged in each category.

A different, but related, set of release fractions were used for barge shipments of N Reactor SNF. The relationship deals with the potential involvement of multiple shipping casks in a barge carrying 24 of them. It is overly conservative to assume that all 24 shipping casks would fail in minor barge accidents. In the lower severity categories, the accident conditions are not severe enough to damage all 24 shipping casks. In fact, in the lowest severity category that results in a release, only the shipping casks in the vicinity of the collision would be affected. Consequently, the release fraction for severity category 2 was multiplied by 1/24 to reflect the assumption that only one of the total of 24 shipping casks aboard the barge would be damaged. Category 3 release fractions were multiplied by 1/12 to reflect the assumption that two shipping casks out of 24 would be damaged in the accident. The release fractions for severity categories 4, 5, and 6 were multiplied by 1/6, 1/3, and 1 to reflect the assumption that 4, 8, and all 24 casks would be damaged, respectively.

Release fractions for HLW shipments were assumed to be the same as those for SNF shipments. The difference is that the strength and durability of the vitrified HLW form was taken into account by assuming that not all of the materials released are in respirable or dispersable form. RADTRAN 4 default values for "immobilized" radionuclides were used to model the dispersable and respirable fractions of the released material. This means that the fraction of released material that is in dispersable form is 1.0E-06, and the respirable fraction is 5.0E-02 (Neuhauser and Kanipe 1992). The HLW release fractions for barge shipments were adjusted similarly to those for SNF to account for the fraction of casks that were assumed to be damaged in the six severity categories.

For plutonium and uranium oxide shipments, no data were readily available. Therefore, the release fractions presented in Table B-7 are representative approximations. It was assumed that 10% of the material released from the plutonium and uranium shipment accidents is in dispersable form and 5% of that is in respirable form, based on recommendations made by Neuhauser and Kanipe (1992) for shipment of small powder materials.

B.4.2.2.2 Consequences of Maximum Reasonably Foreseeable Accidents. The dose to the maximum individual and the collective population dose from the maximum reasonably foreseeable accident was calculated for each type of shipment, i.e., SNF, solidified HLW, and plutonium and uranium oxide. The quantity and radiological constituents of each waste form are discussed in Chapter 2.0 of this appendix. The computer code RISKIND (Yuan et al. 1993) was used to calculate the dose to the maximum individual and the population.

RISKIND Input Parameters. This analysis evaluates the consequences of accidents involving truck or rail shipments. A separate assessment was not performed for barge shipments to Portland because of the similarity between the rail and barge routing data (see Section B.2.3). The radiological inventories developed in Table B-1 have been used to calculate the dose to the maximum individual and the public. For all analyses, inhalation doses were calculated for each of the NRC modal study severity categories, assuming the maximum individual was located 100 m from the point of release and neutral weather conditions (i.e., Atmospheric Stability Class = D and 4 m/s wind speed). To determine the maximum individual dose for each of the material types, the calculated dose for each of the NRC modal study categories (20) were binned into the accident severity categories shown in Table B-6. The results of the RISKIND calculations for each severity category are presented in Table B-8.

An accident frequency (accidents per year) and probable accident location by population zone (i.e., rural, suburban, and urban) were developed for each campaign, based on the type of material, transportation mode, transportation routing information, and state-specific transportation accident data. For this analysis a campaign is defined as the total number of shipments required to transport all of the material from the point of origin to the destination.

For each of the transportation modes, existing transportation model computer codes, i.e., HIGHWAY (Johnson 1993a; population data revised in 1994) and INTERLINE (Johnson 1993b; population data revised in 1994) were used to develop the route-specific information required for the accident analyses.

The information required to calculate the accident frequencies included the total number of shipments per campaign, the campaign duration, the total shipping distance, population zone-specific accident rates by state, and the conditional probabilities shown in Table B-6. The population zone-specific accident frequencies are calculated using the state-specific accident data (accidents per kilometer) for each of the population zones contained in Saricks and Kvitek (1994) and the distance traveled in each of the population zones. The resulting adjusted accident rates are shown in Table B-9. The values in this table were used to select the maximum reasonably foreseeable accident scenario.

Table B-8. RISKIND calculated doses summarized by severity category[a].

Severity Category[b]	Truck: Spent Nuclear Fuel (rem)	Truck: Pu Oxide (rem)	Truck: U Oxide (rem)[c]	Rail: Spent Nuclear Fuel (rem)	Rail: Solidified HLW[d] (rem)
1[e]	2.36E-05	2.36E-05	2.36E-05	2.36E-05	2.36E-05
2	8.59E-03	3.91E-04	2.36E-05	1.30E-01	1.26E-01
3	5.01E-02	1.25E-03	2.36E-05	8.53E-01	8.39E-01
4	9.39E-02	1.23E-02	2.36E-05	2.96E-01	1.26E-01
5	1.18E-01	1.23E-02	2.36E-05	9.80E-01	8.39E-01
6	**2.60E-01**	**1.23E-01**	**2.36E-05**	**1.27E+00**	**8.39E-01**

a. Maximum individual doses are in **BOLD**. (These doses were estimated in the event an accident occurs; i.e., they were not multiplied by the corresponding accident frequencies).
b. Severity categories are defined in Table B-6.
c. Only external doses were calculated.
d. The quantity of HLW released has been adjusted because of the immobilized form of the material. The adjustment, 1.0E-06, was taken from RADTRAN 4 (Neuhauser and Kanipe 1992).
e. Although, no material would be released, an external dose is calculated as a result of changes in the cask shielding caused by an accident impact.

The calculated maximum individual doses were cross referenced with the accident frequencies in Table B-9, and the maximum individual doses for reasonably foreseeable accidents (i.e., the accident frequency is greater than 1 x 10^{-7}/year) have been reported.

The population dose from the maximum reasonably foreseeable accident is also provided. These analyses are based on the same assumptions used to calculate the dose to the maximally exposed individual. The location of the accident (or population zone) is the same as the accident location used to calculate the maximum individual doses. The population densities for each of the impacted population zones were developed using HIGHWAY (Johnson 1993a) and INTERLINE (Johnson 1993b).

Table B-9. Summary of route-specific accident rates.

Total distance (km)	Distance per zone (km)			Travel fraction			Population zone accident rate (1.0E-07/km)		
	Rural	Suburban	Urban	Rural	Suburban	Urban	Rural	Suburban	Urban
Norfolk to Hanford - Truck									
4311.43	3640.28	619.48	51.67	0.84	0.14	0.01	2.508	3.369	4.129
Portland to Hanford -Truck									
416.82	353.25	50.21	13.36	0.85	0.12	0.03	2.279	2.802	3.675
Seattle to Hanford - Truck									
276.80	243.80	27.70	5.30	0.88	0.10	0.02	2.500	2.055	1.610
Norfolk to Hanford - Rail									
4984.78	4140.40	723.60	120.78	0.83	0.15	0.02	0.524	0.678	0.753
Portland to Hanford -Rail									
430.50	366.32	4921	14.97	0.86	0.11	0.03	0.361	0.298	0.271
Seattle to Hanford - Rail									
715.8	530.5	136.4	48.9	0.74	0.19	0.07	0.349	0.349	0.349

B.4.2.3 Results of Transportation Accident Impact Calculations. The results of the integrated population risk assessment are presented in Table B-10. The lowest impact option is that in which SNF is shipped from Hanford to the Port of Seattle by rail. The Port of Seattle by truck option is the next highest followed in order by the rail option to Norfolk, truck to Norfolk, and then barge to Portland. The impacts for all of the options are dominated by the SNF shipments to the U.K. and plutonium oxide return shipments to Hanford, primarily because the quantities and forms of these materials are more vulnerable to accidental releases and represent higher radiotoxicities than vitrified HLW and uranium oxide. Shipments of vitrified HLW were determined to present the lowest impacts of all the materials because of the reasons given plus the immobilized form of the material relative to the other materials.

Shipments by barge are shown in Table B-10 to result in relatively higher accident impacts than shipments by rail or truck. This is because the inventories of radioactive materials transported by barge, and the resulting potential accident releases, are at least an order of magnitude greater than for truck and rail shipments. Because the accident rates for the three modes are comparable, this results in a higher per shipment (or per-km) accident risk for barge than the other modes. This higher per-shipment risk more than offsets the risk reduction attributable to fewer barge

Table B-10. Results of transportation accident risk assessment[a].

Option and material	Accident impacts, person-rem	Latent cancer fatalities
Barge to Portland		
SNF	1.8E-02	9.0E-06
HLW	1.5E-08	7.5E-12
Pu	9.3E-03	4.7E-06
U	2.7E-06	1.4E-09
TOTAL	2.7E-02	1.4E-05
Truck to Seattle		
SNF	9.3E-05	4.7E-08
HLW (Rail)	1.6E-10	8.0E-14
Pu (Truck)	3.6E-03	1.8E-06
U (Truck)	1.1E-06	5.5E-10
TOTAL	3.7E-03	1.9E-06
Rail to Seattle		
SNF	6.3E-05	3.2E-08
HLW (Rail)	1.6E-10	8.0E-14
Pu (Truck)	3.6E-03	1.8E-06
U (Truck)	1.1E-06	5.5E-10
TOTAL	3.7E-03	1.8E-06
Truck to Norfolk		
SNF	2.1E-03	1.1E-06
HLW (Rail)	9.3E-10	4.7E-13
Pu (Truck)	8.3E-02	4.1E-05
U (Truck)	2.4E-05	1.2E-08
TOTAL	8.5E-02	4.2E-05
Rail to Norfolk		
SNF	7.4E-04	3.7E-07
HLW (Rail)	9.3E-10	4.7E-13
Pu (Truck)	8.3E-02	4.1E-05
U (Truck)	2.4E-05	1.2E-08
TOTAL	8.3E-02	4.2E-05

a. Reported values are point estimates of risk; i.e., the accident frequency multiplied by the consequences that would be expected if an accident occurred.

shipments so, overall, barge accident risks appear to be higher than truck or rail transport risks. However, in comparing the magnitudes of the accident risks in Table B-8 to the public routine exposures in Table B-5, it can be seen that the accident risks are lower than the routine public exposures. Consequently, it may be concluded that transportation accident risk impacts are insignificant contributors to the total impacts of the transportation options.

The results of the maximum reasonably foreseeable accident consequence assessment are provided in Tables B-11 through B-14. The results in these tables were generated using the RISKIND computer code. The following paragraphs discuss the results of the maximally exposed individual consequence assessment for each material. This is followed by a discussion of the results of the collective dose calculations.

N Reactor SNF. As discussed in Section 2.0, SNF will be loaded into shipping casks at the K Basins and transported by barge, truck, or rail to ocean ports for shipment to the U.K. Two shipping modes and three transportation routes were evaluated. The radiological source inventory used in the analysis was shown in Table B-1. The release fractions used here were taken from Volume 1, Appendix I of this EIS (see Table B-7). The results of the evaluation are shown in Table B-11.

As can be seen in Table B-11, for reasonably foreseeable events (i.e., the accident frequency is greater than 1.0E-07/year), the dose received by the maximally exposed individual from a rail accident ranges from 9.80E-01 to 1.27E+00 rem depending on the location of the individual and transportation route. The potential LCF range from 4.90E-04 to 6.35E-04. The accident frequency also varies based on the transportation route and accident location from 1.27E-07 to 1.91E-06/year. Table B-11 also presents the dose received by the maximally exposed individual from a truck accident. The dose to the maximally exposed individual ranges from 1.18E-01 to 2.60E-01 rem, depending on the location of the individual and transportation route. The accident frequency also varies based on the transportation route and accident location from 1.23E-07 to 1.02E-05/year. The potential LCF range from 5.90E-05 to 1.30E-04.

Collective doses to the public were also calculated for each of the transport modes and transportation route (see Table B-11). For this analysis, it was assumed that the accident occurred in the same location as that determined in the maximum individual dose calculations. The population dose from a rail accident ranges from 3.18E+00 to 3.27E+02 person-rem depending on the accident location, population density, and transportation route. The doses to population from a truck accident range from 1.37E-01 to 9.44E+02 person-rem. The potential LCF range from 1.59E-03 to 0.170 for rail and 6.85E-05 to 4.72E-1 for truck.

Table B-11. Calculated maximum individual and population radiological doses and latent cancer fatalities based on accident location and frequency of SNF shipments.

Transportation Route	Mode	No. of ship-ments[a]	Accident frequency (per year)[b]	Accident location: population zone[c]	Maximum individual TEDE[d] (rem)	Maximum individual LCF[e]	Population TEDE[d] (person-rem)	Population LCF[e]
Hanford, Washington to Portland, Oregon	Truck	408	1.23E-07	Urban	2.60E-01	1.30E-04	1.01E+02	5.05E-02
Hanford, Washington to Seattle, Washington			1.02E-05	Rural	1.18E-01	5.90E-05	1.37E-01	6.85E-05
Hanford, Washington to Norfolk, Virginia			1.43-06	Urban	2.60E-01	1.30E-04	9.44E+02	4.72E-01
Hanford, Washington to Portland, Oregon	Rail	204	3.46E-07	Rural	9.80E-01	4.90E-04	3.18E+00	1.59E-03
Hanford, Washington to Seattle, Washington			1.27E-07	Urban	1.27E+00	6.35E-04	3.39E+02	0.170
Hanford, Washington to Norfolk, Virginia			1.91E-06	Urban	1.27E+00	6.35E-04	3.27E+02	0.164

a. Assumes one truck cask per truck shipment and two truck casks per rail shipment.

b. Accident frequency based on the number of shipments, campaign duration, one-way shipping distance, and conditional probability.

c. Accident location is based on population zone where the maximum individual dose occurs.

d. TEDE - 50-year total effective dose equivalent.

e. LCF - Latent cancer fatalities. Calculated on dose (rem) to maximum individual or population, i.e., 5.0E-04 LCF/rem

Table B-12. Calculated maximum individual and population radiological doses and latent cancer fatalities based on accident location and frequency for plutonium oxide shipments.

Transportation Route	Mode	No. of Ship.[a]	Accident Frequency (per year)[b]	Accident Location: Population Zone[c]	Maximum Individual TEDE[d] (rem)	Maximum Individual LCFs[e]	Population TEDE[d] (rem)	Population LCFs[e]
Portland, Oregon to Hanford, Washington	Truck	186	1.22E-07	Urban	1.23E-01	6.15E-05	1.88E+01	9.40E-03
Seattle, Washington to Hanford, Washington			1.01E-05	Rural	1.23E-02	6.15E-06	3.46E-03	1.73E-06
Norfolk, Virginia to Hanford, Washington			1.42E-06	Urban	1.23E-01	6.15E-05	1.77E+01	8.85E-03

a. Assumes one cask per truck shipment.

b. Accident frequency based on the number of shipments, campaign duration, one-way shipping distance, and conditional probability.

c. Accident location is based on population zone where maximum individual dose occurs.

d. TEDE - 50 year Total Effective Dose Equivalent.

e. LCFs - Latent cancer fatalities. Calculated based on dose (rem) to maximum individual or population, i.e., 5.0E-04 LCFs/rem

Plutonium Oxide. The separated plutonium oxide was assumed to be returned to its point of origin (i.e., Hanford). This material was assumed to be transported to a U.S. port (Seattle, Portland, or Norfolk) by ocean-going ship and offloaded to a Safe-Secure Trailer/Armored Tractor for subsequent highway shipment to Hanford (one container per shipment).

The results of this analysis are provided in Table B-12. The dose, to the maximally exposed individual from the maximum reasonable foreseeable accident, ranges from 1.23E-02 to 1.23E-01 rem, depending on the location of the individual and transportation route. The potential LCF ranges from 5.90E-06 to 5.90E-05. The accident frequency ranges from 1.22E-07 to 1.01E-05/year depending on the transportation route and accident location.

The potential population doses from the maximum reasonably foreseeable accident have also been calculated and are shown in Table B-12. Assuming that the accident occurs in the same location or population zone as that determined for the maximally exposed individual, the population dose ranges from 3.46E-03 to 1.88E+01 person-rem. The potential LCF range from 1.73E-06 to 9.40E-03.

Uranium Oxide. As with plutonium oxide, uranium oxide resulting from SNF processing was assumed to be returned to Hanford. This material was assumed to be transported by ship to a port facility where it would be offloaded onto a truck for subsequent highway transport to Hanford. As with the plutonium oxide, only truck accidents were evaluated. The calculated dose received by the maximum individual from a truck accident is 2.36E-05 rem (see Table B-13). The potential LCF are 1.18E-08. The accident frequency ranges from 1.23E-07 to 1.01E-05 per year depending on the transportation route and accident location.

The potential collective dose ranges from 3.65E-06 to 1.98E-03 person-rem depending on the location and transportation route. The potential LCF range from 1.83E-09 to 9.90E-07 and also depend on the accident location and transportation route.

Solidified High-Level Waste. Following separation of all plutonium and uranium from the N Reactor fuel, the resulting HLW was assumed to be vitrified and poured into canisters. These canisters were assumed to be shipped in rail shipping casks by ship to a U.S. port facility and offloaded to rail cars at the port; therefore, only rail accidents were evaluated for shipments of HLW. The radiological source inventory used in the analysis was shown in Table B-1 and the release fractions were shown in Table B-7. Because the waste material that has been solidified in

Table B.13. Calculated maximum individual and population radiological doses and latent cancer fatalities based on accident location and frequency for uranium oxide shipments.

Transportation route	Mode	No. of ship-ments[a]	Accident frequency (per year)[b]	Accident location: population zone[c]	Maximum individual TEDE[d] (rem)	Maximum individual LCF[e]	Population TEDE[d] (person-rem)	Population LCF[e]
Portland, Oregon to Hanford, Washington	Truck	236	1.23E-07	Urban	2.36E-05	1.18E-08	1.98E-03	9.90E-07
Seattle, Washington to Hanford, Washington			1.01E-05	Rural	2.36E-05	1.18E-08	3.65E-06	1.83E-09
Norfolk, Virginia to Hanford, Washington			1.43E-06	Urban	2.36E-05	1.18E-08	1.86E-03	9.3E-07

a. Assumes one cask per truck shipment.
b. Accident frequency based on the number of shipments, campaign duration, one-way shipping distance, and conditional probability.
c. Accident location is based on the population zone where maximum individual dose occurs.
d. TEDE - 50-year total effective dose equivalent.
e. LCF - Latent cancer fatalities. Calculated on dose (rem) to maximum individual or population, i.e., 5.0E-04 LCF/rem.

glass logs was considered to be "immobilized" material, the fraction of released material that is also dispersable and the fraction that is also respirable were adjusted, as discussed in Section 4.2.2.1.

The calculated dose to the maximally exposed individual and population are shown in Table B-14. The dose to the maximally exposed individual was 8.39E-01 rem and the potential latent cancer fatalities would be 4.20E-04. The accident frequency varies by route and ranges from 1.25E-07 to 1.88E-06/year.

The population doses are also shown in Table B-14. The collective dose ranges from 3.48E+00 to 1.42E+03 person-rem. The potential latent cancer fatalities range from 1.74E-03 to 0.710.

B.4.2.4 Assessment of Nonradiological Impacts. Nonradiological accident impacts consist of fatalities that may result from traffic accidents involving the shipments to and from the offshore processing facility. Nonradiological incident-free impacts are those resulting pollutants emitted from the vehicles. These impacts are not related to the radioactive nature of the materials being transported. In fact, the number of estimated injuries and fatalities would be the same even if the cargo were not radioactive materials. This section uses unit risk factors to estimate the nonradiological impacts associated with the five shipping scenarios considered in this evaluation.

The potential for accidents involving shipments of materials to and from an offshore processing facility is assumed to be comparable to that of general truck, rail, and barge transport in the U.S. Nonradiological accident unit risk factors were taken from Saricks and Kvitek (1994) to calculate nonradiological accident impacts. These risk factors, in units of fatalities-per-km of travel in rural and urban population zones, were multiplied by the total distance traveled in each zone by all of the shipments and then summed to calculate the expected number of nonradiological fatalities. The unit risk factor for travel in suburban zones was represented by the average of the rural and urban unit risk factors given by Saricks and Kvitek (1994).

Impacts to the public from non-radiological causes are also evaluated. This includes fatalities resulting from pollutants emitted from the vehicles during normal transportation. Based on the information contained in Rao et al. (1982), the types of pollutants that are present and can impact the public are sulfur oxides (SO_x), particulates, nitrogen oxides (NO_x), carbon monoxide (CO), hydrocarbons (HC), and photochemical oxidants (O_x). Of these pollutants, Rao et al. (1982) determined that the majority of the health effects are from SO_x and the particulates. Unit risk

Table B-14. Calculated maximum individual and population radiological doses and latent cancer fatalities based on accident location and frequency for solidified high level waste shipments

Transportation Route	Mode	No. of ship-ments.[a]	Accident frequency (per year)[b]	Accident location: population zone[c]	Maximum individual TEDE[d] (rem)	Maximum individual LCF[e]	Population TEDE[d] (person-rem)	Population LCF[e]
Portland, Oregon to Hanford, Washington	Rail	24	3.39E-07	Rural	8.39E-01	4.20E-04	3.48E+00	1.74E-03
Seattle, Washington to Hanford, Washington			1.25E-07	Urban	8.39E-01	4.20E-04	1.42E+03	7.1E-01
Norfolk, Virginia to Hanford, Washington			1.88E-06	Urban	8.39E-01	4.20E-04	1.37E+03	6.8E-01

a. Assumes one cask per rail shipment.

b. Accident frequency based on the number of shipments, campaign duration, one-way shipping distance, and conditional probability.

c. Accident location is based on population zone where maximum individual dose occurs.

d. TEDE - 50-year total effective dose equivalent.

e. LCF - Latent cancer fatalities. Calculated on dose (rem) to the maximum individual or population, i.e., 5.0E-04 LCF/rem.

factors (fatalities per kilometer) for both truck and rail shipments were developed by Rao et al. (1982) for travel in urban population zones (1.0E-07/km and 1.3E-07/km truck and rail respectively). These unit risk factors were combined with the total shipping distance in urban population zones to calculate the nonradiological incident-free impacts to the public.

The results of the nonradiological accident and incident-free impact calculations for the five potential shipping scenarios are presented in Table B.15. The values reported in the table represent the sum of the impacts from all of the shipments and include the impacts from shipments carrying cargo as well as those from empty return shipments.

B.4.3 Dose to the Public from Port Activities

Normal port activities during transport of N Reactor SNF are not expected to have any consequences for members of the public other than port workers, as discussed in Section 3.3.

The consequences of accidents during port transit were estimated using the same assumptions described for worker consequences in Section 3.3.2. Collective point estimates of risk to the population within 50 miles (80 km) of each location was estimated for an accident at the dock and on the approach to the port. The point estimate of risk to an individual at 1600 m (1 mile) was also estimated for applicable exposure pathways as described in Attachment A of this appendix. Consequences for populations and individuals are reported, both with and without the risk from ingestion of locally grown foods because protective action guidelines would require mitigative actions if the projected dose exceeded specified levels. Individual consequences assume 95% atmospheric dispersion, whereas consequences to populations are estimated for both 50% and 95% atmospheric dispersion.

Table B.15. Nonradiological transportation impacts of offshore processing scenarios

Shipping scenario	Accident impacts, fatalities	Incident-free impacts, fatalities
Barge to Portland	1.1E-02	2.1E-03
Seattle by Truck	8.9E-03	1.2E-03
Seattle by Rail	1.2E-02	3.4E-03
Norfolk by Truck	1.3E-01	1.6E-02
Norfolk by Rail	1.2E-01	1.5E-02

The consequences of port accidents were estimated in a manner similar to that used for overland transportation impacts. The contents of one shipping cask were assumed to be involved in an accident (see Table B-1), with radionuclide releases according to the release fractions reported in Table B-7. The dose and resulting LCF were calculated for each of the six accident severity categories. The point estimates of risk included the consequences as LCF for accidents of each severity category multiplied by the frequency with which an accident of that severity would occur. The accident frequencies for each severity category were assumed to be the overall accident rate per port transit (3.2×10^{-4}) multiplied by the conditional probability for accidents in each severity category listed in Table B-6 (DOE 1994). The total accident risk for an individual or population was then estimated as the sum of risks for all accident severity categories. Risks for accidents evaluated at 95% (stable) atmospheric dispersion were assumed to be 10% lower than those at 50% (neutral) dispersion.

The results for accidents at the four representative ports are shown in Table B-16, with estimated risks for individual residents and populations within 80 km (50 miles). Point estimates of risk for the individual resident ranged from 6.2×10^{-13} to 1.3×10^{-11} LCF if no locally grown food were considered; results for all exposure pathways including ingestion were 3.5×10^{-11} to 7.8×10^{-10} LCF.

Collective point estimates of risk to the population within 50 miles of Portland, Oregon were 5.2×10^{-9} to 4.9×10^{-6} LCF assuming 50% atmospheric dispersion conditions and 1.0×10^{-8} to 8.3×10^{-6} LCF for 95% atmospheric dispersion. Corresponding results for the population in the vicinity of Newark are 2.3×10^{-8} to 4.9×10^{-5} LCF assuming 50% atmospheric dispersion and 1.5×10^{-8} to 8.4×10^{-5} LCF for 95% atmospheric dispersion. Consequences for the collective populations of Seattle-Tacoma and Norfolk fell between the estimates for the other two ports.

The maximum reasonably foreseeable accident was a category 6 accident, which has a frequency of 1.3×10^{-7} per port transit, and which was evaluated for either neutral or stable atmospheric conditions resulting in a cumulative frequency of 2.2×10^{-6} or 2.2×10^{-7}, respectively for 17 SNF shipments. Dose and risk estimates for the maximum reasonably foreseeable accident are presented in Table B-17. The dose to the resident member of the public ranged from an estimated 0.02 to somewhat over 1 rem for all ports, depending on whether locally grown food was considered as an exposure pathway. The corresponding probability of LCF ranged from 9.0×10^{-6} to 6.5×10^{-4} and point estimates of risk, from 2.0×10^{-12} to 1.4×10^{-10} LCF. The collective

Table B-16. Point estimate of risk[a] of latent cancer fatalities from port accidents.

Port location	Portland, Oregon		Seattle-Tacoma, Washington		Norfolk, Virginia		Newark, New Jersey	
Exposure Pathways	All pathways	Inhalation + external	All pathways	Inhalation + external	All pathways	Inhalation + external	All pathways	Inhalation + external
Individual at 1600 m - 95% (stable) atmospheric conditions								
1 Shipment	4.6E-11	7.9E-13	3.5E-11	6.2E-13	4.6E-11	7.9E-13	3.9E-11	6.8E-13
17 Shipments	7.8E-10	1.3E-11	6.0E-10	1.0E-11	7.8E-10	1.3E-11	6.7E-10	1.2E-11
Population within 80 km (50 miles) of dock - 50% (neutral) atmospheric conditions								
1 Shipment	2.9E-07	6.6E-09	1.9E-07	4.3E-09	1.2E-07	2.7E-09	1.0E-06	2.3E-08
17 Shipments	4.9E-06	1.1E-07	3.2E-06	7.2E-08	2.0E-06	4.6E-08	1.7E-05	3.9E-07
Population within 80 km (50 miles) of harbor approach - 50% (neutral) atmospheric conditions								
1 Shipment	2.4E-07	5.2E-09	6.0E-08	1.4E-09	1.1E-07	2.5E-09	2.9E-06	6.5E-08
17 Shipments	4.0E-06	8.9E-08	1.0E-06	2.3E-08	1.9E-06	4.3E-08	4.9E-05	1.1E-06
Population within 80 km (50 miles) of dock - 95% (stable) atmospheric conditions								
1 Shipment	4.5E-07	1.0E-08	2.3E-07	5.1E-09	3.3E-07	7.4E-09	5.0E-06	1.5E-08
17 Shipments	7.6E-06	1.8E-07	3.9E-06	8.8E-08	5.6E-06	1.3E-07	8.4E-05	2.5E-07
Population within 80 km (50 Miles) of Harbor Approach - 95% (stable) Atmospheric Conditions								
1 Shipment	4.9E-07	1.0E-08	1.2E-07	2.8E-09	2.5E-07	5.8E-09	4.9E-06	1.1E-07
17 Shipments	8.3E-06	1.7E-07	2.0E-06	4.7E-08	4.3E-06	9.8E-08	8.3E-05	1.9E-06

a. Point estimate of risk is defined as the consequences to the receptor or population (as LCF) of an accident of a given severity category (assuming the accident occurs), multiplied by the frequency per shipment with which an accident of that severity would occur. The risks for accidents of all severity categories are then summed to obtain the total risk per shipment.

consequences to the populations within 80 km (50 mi) of the ports ranged from 2.0 x 10^{-3} to 380 LCF assuming the accident occurs, depending on the location of the accident (port or harbor approach) and the exposure pathways considered. The corresponding point estimates of risk for latent fatal cancers amounted to 4.4 x 10^{-9} to 8.2 x 10^{-5}.

B.4.4 Dose to the Public from Ocean Transport to the United Kingdom

This analysis expects no dose to members of the public resulting from incident-free ocean transport of N Reactor SNF to the U.K. The ships carrying the fuel are owned and operated by the commercial vendor, and its shipboard crews are assumed to be classified as radiation workers for the purposes of this analysis.

The effects of losing a cask at sea are estimated to be comparable to those evaluated for shipment of foreign research reactor SNF to the U.S. (DOE 1994), based on similar shipping inventories of long-lived radionuclides per cask. The maximum dose to an individual for a cask lost in coastal waters was expected to be 11 mrem/year if the cask were left in place until all its contents dispersed. The corresponding consequences to marine biota were 0.24 mrad/year for fish, 0.32 mrad/year for crustaceans, and 13 mrad/year for mollusks. The consequences resulting from loss of a cask in the deep ocean would be many orders of magnitude lower than estimates for coastal waters.

The probability of accident on the open ocean was estimated to be 4.6 x 10^{-5} per shipment for an average duration voyage of about 20 days in transporting SNF from foreign research reactors to the U.S. (DOE 1995). The frequency of accidents for overseas shipment of SNF and process materials via special-purpose ships would likely be within a factor of two or three of this estimate. However, that frequency applies to commercial freight shipping experience, and it is possible that the use of special-purpose ships could result in a different accident rate. Using the commercial freight accident rate given above, the probability of an accident on the open ocean involving transport of SNF (17 ocean shipments), HLW (1 shipment), uranium oxide (1 shipment), and plutonium oxide (1 shipment) was calculated to be about 9.2E-04, integrated over all the shipments.

Table B-17. Consequences and risk to the public surrounding port facilities from maximum reasonably foreseeable accidents involving SNF shipments at or near the ports

Port Location	Portland, Oregon		Tacoma, Washington		Norfolk, Virginia		Newark, New Jersey	
	All pathways	Inhalation + external	All pathways	Inhalation + External	All pathways	Inhalation + eternal	All pathways	Inhalation + external
Resident at 1600 m								
Dose (rem)	1.3E+00	2.3E-02	9.9E-01	1.8E-02	1.3E+00	2.3E-02	1.1E+00	2.0E-02
LCF	6.5E-04	1.2E-05	5.0E-04	9.0E-06	6.5E-04	1.2E-05	5.5E-04	9.9E-06
LCF risk	1.4E-10	2.5E-12	1.1E-10	2.0E-12	1.4E-10	2.5E-12	1.2E-10	2.2E-12
Population within 80 km (50 mi) of dock - 50% (neutral) atmospheric dispersion								
Dose (person-rem)	8.7E+02	1.9E+01	5.5E+02	1.2E+01	3.5E+02	7.7E+00	3.1E+03	6.8E+01
LCF	4.4E-01	9.7E-03	2.8E-01	6.0E-03	1.8E-01	3.9E-03	1.6E+00	3.4E-02
LCF risk	9.5E-07	2.1E-08	6.0E-07	1.3E-08	3.8E-07	8.4E-09	3.4E-06	7.3E-08
Population within 80 km (50 mi) of harbor approach - 50% (neutral) atmospheric dispersion								
Dose (person-rem)	6.9E+02	1.5E+01	1.8E+02	4.0E+00	3.3E+02	7.3E+00	8.5E+03	1.8E+02
LCF	3.5E-01	7.5E-03	9.0E-02	2.0E-03	1.7E-01	3.7E-03	4.3E+00	9.1E-02
LCF risk	7.5E-07	1.6E-08	2.0E-07	4.4E-09	3.6E-07	7.9E-09	9.2E-06	2.0E-07
Population within 80 km (50 mi) of dock - 95% (stable) atmospheric dispersion								
Dose (person-rem)	1.3E+04	2.9E+02	6.9E+03	1.5E+02	9.8E+03	2.1E+02	7.5E+05	1.7E+03
LCF	6.5E+00	1.4E-01	3.5E+00	7.5E-02	4.9E+00	1.1E-01	3.8E+02	8.6E-01
LCF risk	1.4E-06	3.1E-08	7.5E-07	1.6E-08	1.1E-06	2.3E-08	8.2E-05	1.9E-07
Population within 80 km (50 mi) of harbor approach - 95% (stable) atmospheric dispersion								
Dose (person-rem)	1.4E+04	3.1E+02	3.6E+03	7.8E+01	7.5E+03	1.6E+02	1.4E+05	3.2E+03
LCF	7.0E+00	1.6E-01	1.8E+00	3.9E-02	3.8E+00	8.0E-02	7.0E+01	1.6E+00
LCF risk	1.5E-06	3.4E-08	3.9E-07	8.5E-09	8.2E-07	1.7E-08	1.5E-05	3.5E-07

B.5 Legal and Policy Considerations

B.5.1 Policy Considerations

For a general discussion of the policy considerations associated with DOE's management of SNF, see Section 2 of Volume 1. Several policy considerations bear on the evaluation of international shipment and processing of SNF.

The primary consideration in international shipment of nuclear materials is concern for unauthorized diversion of such materials to foreign weapons programs (nuclear proliferation). This concern is mitigated, but not eliminated, because SNF is not directly useable in simple nuclear weapons. Stringent safeguards exist for overseas transportation of nuclear materials. Highly enriched uranium has been transported overseas for research purposes, and SNF from research reactors has been returned to the U.S. for disposition. Although such return shipments have not occurred routinely since 1988, DOE is considering resumption of such shipments in support of U.S. efforts to remove highly enriched uranium SNF from international commerce. Two such shipments were completed on an urgent relief basis in 1994, and additional shipments may resume on completion of an evaluation by DOE (1995).

DOE (1993) has evaluated the safety and policy issues associated with overseas transport of plutonium and concluded that such shipments could be made safely and securely within the context of current national and international regulations for transport of radioactive materials (including special nuclear materials). The report (DOE 1993) addresses risks to the public and the environment, emergency response requirements, safeguards, and the regulatory framework within which such shipments could be made.

The overseas transportation of SNF and eventual return of vitrified wastes and end products contemplated in this alternative would be managed in accordance with well defined and demonstrated practices. However, a decision to implement the overseas transportation and processing option will require close examination of various policy and international documents that address plutonium stockpiling and the exchange of nuclear materials.

Other major policy considerations are the comparative risk of overseas shipment and return versus strictly domestic transportation and management of SNF and the involvement of a foreign population and environment in the foreign processing alternative. A decision to implement the BNFL option would be likely to generate controversy over the perception of transferrring environmental problems overseas. Transportation risks are addressed in Sections B.3 and B.4 of this attachment.

The representative facility used for this analysis (British Nuclear Fuels facility operations in Sellafield, U.K.) began in the 1940s with the same primary mission as Hanford. This commercial facility processes large volumes of SNF from several foreign countries. Round trip shipments and management of SNF and waste products would therefore be undertaken within a demonstrated regulatory, technical, and physical infrastructure.

B.5.2 Applicable Laws, Regulations, and Other Requirements

B.5.2.1 General. This discussion is limited to regulatory considerations associated with the round trip domestic and overseas transportation of SNF and other hazardous and radioactive materials. For a discussion of general laws and regulation governing the management of SNF, see Section 2.2 of this appendix. State and local requirements will not be discussed here because the shipments of SNF under consideration would be in interstate or foreign commerce and federal provisions would govern. Internal DOE Orders also are not discussed.

The significant international and federal laws and regulations that apply to the transportation of hazardous and radioactive materials include the following laws:

- International Convention on the Safety of Life at Sea of 1960 (as amended)
- Atomic Energy Act (42 U.S.C. 2011 et seq.)
- Hazardous Transportation Materials Act (49 U.S.C. 1801 et seq.)
- Resource Conservation and Recovery Act, as amended by the Hazardous and Solid Waste Amendments (42 U.S.C. 26901 et seq.)
- Executive Order 12898 (Environmental Justice)
- Executive Order 12114 (Environmental Effects Abroad of Major Federal Actions).

B.5.2.2 Domestic Packaging and Transportation. Transportation of hazardous and radioactive materials, substances, and wastes are governed by the regulations of the U.S. Department of Transportation (DOT) (49 CFR 171-178, 383-397), the U.S. Nuclear Regulatory Commission (NRC) (10 CFR 71), and the U.S. Environmental Protection Agency (EPA) (40 CFR 262, 265).

United States DOT regulations contain requirements for identifying a material as hazardous or radioactive. These regulations interface with NRC and EPA regulations for identifying material, but the DOT regulations govern hazard communication via placarding, labeling, reporting, and shipping requirements (see especially 10 CFR 71.5, in which DOT regulations are applied to shipping of radioactive materials by NRC regulations).

Nuclear Regulatory Commission regulations address packaging design and certification requirements. Certification is based on safety analysis report data on the packaging design for various hypothetical accident conditions.

General overland carriage is governed by specific regulations dealing with packaging notification, escorts, and communication. There are specific provisions for truck and for rail. For carriage by truck, the carrier must use interstate highways or state-designated preferred routes. Department of Transportation regulations found in 49 CFR 397.101 establish routing and driver training requirements for highway carriers of packages containing "highway-route-controlled quantities" of radioactive materials. Spent nuclear fuel shipments constitute such controlled shipments. For carriage by rail car, each shipment by the railroad must comply with 49 CFR 174 Subpart K "Detailed Requirements for Radioactive Materials."

B.5.2.3 Overseas Transportation. To the extent feasible, the NRC and DOT conform their regulations to the model regulations of the International Atomic Energy Agency. These model international regulations are also incorporated into the International Maritime Dangerous Goods Code, which was developed to supplement the International Convention on the Safety of Life at Sea, to which the U.S. is a signatory. Transportation risk in the global commons must be evaluated in accordance with Executive Order 12114 (Environmental Effects Abroad of Major Federal Actions).

Transportation of dangerous cargoes through the Panama Canal is governed by the International Maritime Dangerous Goods Code (IMDG) and is addressed in 35 U.S.C. 113.

General provisions for passage through the Panama Canal are found at 35 U.S.C. 101-135. General regulations governing navigation, including the applicability of the International Regulations for the Prevention of Collisions at Sea (1972), are found throughout Title 33 of the CFRs.

Relevant regulations applying to transport of SNF by vessel are found in 10 CFR Parts 71 and 73 (NRC) and 49 CFR Part 176 (DOT). These regulations address prenotification to the U.S. Coast Guard for inspection, and provide specifications for packaging, labelling, and other preparation for shipment. A Certification of Competent Authority must be obtained in compliance with International Atomic Energy Agency requirements. Specific provisions are made for stowage, including package surface temperature limitations, spacing, and total aggregate volume and number of freight containers.

B.6 Environmental Justice

For analytical purposes, three modes of transportation were selected for evaluation: 1) truck or rail to a port on Puget Sound (such as Tacoma, Washington); 2) barge to a Columbia River port in the vicinity of Portland, Oregon; or 3) rail or truck across the country to an East Coast port. The East Coast port of reference was assumed to be Norfolk, Virginia (Hampton Roads). These three modes are considered to provide a reasonable range of ports and transportation options for evaluation.

The DOE draft Environmental Impact Statement on the Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor (FRR) Spent Nuclear Fuel (DOE/EIS-0218D) provides information on the numbers and spatial locations of minority and low-income populations surrounding the ports of interest identified above and the Hanford Site. Because the FRR EIS (see Section A.2) utilized somewhat different analytical methodologies for environmental justice purposes than those utilized in this document, some data may vary. The reasons for such variations are explained in Section L-3.5 of Appendix L of this document. Utilizing demographic data entirely from the FRR EIS for the purposes of this attachment, allows for comparison of the sites of interest under consistent definitions and assumptions because the ports identified above were not demographically evaluated in Appendix L of this EIS. The reader is referred to the draft FRR EIS for maps locating the spatial distribution of minority and low income populations.

Table B-18 lists information on selected populations of interest for regions surrounding the Hanford loading facility and ports. Regions surrounding each port are areas that lie at least partially within a 16-km (10-mile) radius of the port. Eighty kilometers (50 miles) is used for Hanford. Population characteristics shown in the table were extracted from detailed, block-group statistical population data of the 1990 census. A block group usually includes 250 to 550 housing units.

Because the impacts as a result of transportation and facility operations are small and reasonably foreseen accidents present no significant risk, no reasonably foreseeable adverse impacts have been identified to the surrounding population. Therefore, no disproportionately high and adverse effects would be expected for any particular segment of the population, including minority and low-income populations.

Table B-18. Characterization of populations residing near candidate facilities (Hanford Site and candidate ports of embarkation[a]).

	Total population within 16 km of facility	Total minority population within 16 km of facility[b]		Households within 16 km of facility	Low income households within 16 km of facility	
Facility	Number	Number	Percent	Number	Number	Percent
Hanford, Washington[c]	383,934	95,042	24.8	136,496	57,667	42.2
Tacoma, Washington	511,575	85,341	16.7	198,458	83,101	41.9
Portland, Oregon	356,064	54,704	15.4	146,047	66,186	45.3
Norfolk, Virginia	681,864	300,179	44.0	206,464	90,723	43.9

a. Data based on draft FRR EIS (DOE/EIS-0218D).
b. Hispanic origin individuals can be of any race.
c. In the case of the Hanford loading facility, a radius of 80 km rather than 16 km was used to define the nearby population.

B.7 Cost

The cost estimate for the foreign processing option, as provided by the representative facility, includes the full service of transporting the SNF from the Hanford Site to the U.K. facility, processing the material into recovered uranium and plutonium and HLW, packaging these products appropriately for return to the U.S., storing the packaged materials pending shipment, and transporting the materials back to the U.S. (BNFL 1994). The proposal provides only a range of total cost ($1.3 - $2 billion), with no breakdown of those costs into the principal cost elements. Thus, there is no detailed estimate of costs for the individual parts of the full service package. The above estimate does not include costs incurred at Hanford to package and stabilize the fuel, if necessary, prior to shipment, or to manage degraded fuel and sludge that may not be suitable for overseas shipment.

B.8 References

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Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement

Volume 1
Appendix B

Idaho National Engineering Laboratory Spent Nuclear Fuel Management Program

April 1995

U.S. Department of Energy
Office of Environmental Management
Idaho Operations Office

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TABLES

FIGURES

1. INTRODUCTION

The U.S. Department of Energy (DOE) has prepared the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (SNF and INEL EIS) to assist its management in making two decisions. The first decision, which is programmatic, is to determine the management program for DOE spent nuclear fuel. The second decision is on the future direction of environmental restoration, waste management, and spent nuclear fuel management activities at the Idaho National Engineering Laboratory.

Volume 1 of the EIS, which supports the programmatic decision, considers the effects of spent nuclear fuel management on the quality of the human and natural environment for planning years 1995 through 2035. DOE has derived the information and analysis results in Volume 1 from several site-specific appendixes. Volume 2 of the EIS, which supports the INEL-specific decision, describes environmental impacts for various environmental restoration, waste management, and spent nuclear fuel management alternatives for planning years 1995 through 2005.

This Appendix B to Volume 1 considers the impacts on the INEL environment of the implementation of various DOE-wide spent nuclear fuel management alternatives. The Naval Nuclear Propulsion Program, which is a joint Navy/DOE program, is responsible for spent naval nuclear fuel examination at the INEL. For this appendix, naval fuel that has been examined at the Naval Reactors Facility and turned over to DOE for storage is termed naval-type fuel. This appendix evaluates the management of DOE spent nuclear fuel including naval-type fuel. Naval spent nuclear fuel examination is addressed in Appendix D; Section 5.16 of this appendix includes relevant environmental consequences from Appendix D.

In addition to this introduction, Appendix B contains the following chapters:

- Chapter 2 - Background: Describes INEL spent nuclear fuel facilities, the regulatory framework for spent nuclear fuel management at the INEL, and the INEL spent nuclear fuel management program.

- Chapter 3 - Spent Nuclear Fuel Management Alternatives: Describes the DOE-wide spent nuclear fuel management alternatives as the INEL would implement them, and provides a summary comparison of potential environmental consequences for each alternative, as described in Chapter 5.

- Chapter 4 - Affected Environment: Describes the INEL site and the surrounding environment that DOE spent nuclear fuel management actions could affect.

- Chapter 5 - Environmental Consequences: Provides the results of environmental consequence analyses for each spent nuclear fuel management alternative.

- Chapter 6 - References

Volume 1 contains a list of acronyms and abbreviations and a glossary that is applicable to this appendix.

2. BACKGROUND

This chapter contains an overview of the Idaho National Engineering Laboratory (INEL) facilities and historic events related to spent nuclear fuel, a description of the regulatory framework for the actions evaluated in this document, and an overview of the current spent nuclear fuel management program at the INEL.

2.1 Overview

The following sections provide a general overview of the INEL including its history, current activities, and mission as they relate to spent nuclear fuel management and future decisions.

2.1.1 History of Spent Nuclear Fuel Activities

The U.S. Atomic Energy Commission, a predecessor of the U.S. Department of Energy (DOE), established the INEL, formerly the National Reactor Testing Station, to build, test, and operate various types of nuclear reactors, support plants, and associated equipment. Since its establishment in 1949 (see Table 2-1), DOE and its predecessor agencies have built 52 reactors at the INEL. The major DOE programs at the site have included test irradiation services, uranium recovery from highly enriched spent fuels, calcination of liquid radioactive waste, light-water-cooled reactor safety testing and research, operation of research reactors, environmental restoration, and storage and surveillance of solid transuranic wastes. In support of the DOE reactor research program and as part of the spent nuclear fuel reprocessing program, the INEL has received spent nuclear fuel from more than 30 offsite sources, including naval reactors, university reactors, commercial reactors, and DOE research reactors, as well as fuels fabricated in the United States and irradiated in foreign reactors (DOE 1993).

The Experimental Breeder Reactor-I, now a National Historic Landmark, maintains a key place in the history of nuclear power in the United States. In December 1951, this reactor generated the first usable electricity from a nuclear reactor. The Experimental Breeder Reactor-I also demonstrated that a nuclear reactor could actually produce more fuel than it consumes.

Of special significance to spent nuclear fuel is the history of the Idaho Chemical Processing Plant. From 1953 to 1992, this plant recovered usable uranium from spent nuclear fuel from United States government reactors. The plant operated for 39 years as a full-scale production facility. But in

Table 2-1. INEL spent nuclear fuel history.

Year	Event
1949	National Reactor Testing Station established
1951	Site reactor first to generate electricity from nuclear fission
1953	ICPP[a] began operation
1953	Test of first submarine nuclear reactor
1957	Expended Core Facility constructed
1965	DOE contract with Public Service Company of Colorado (Fort St. Vrain)
1974	Site became Idaho National Engineering Laboratory
1980	DOE contracted to receive Public Service Company of Colorado (Fort St. Vrain) spent nuclear fuel
1992	Decision to discontinue reprocessing of spent nuclear fuel at ICPP[a] announced
1992	DOE creates Office of Spent Fuel Management
1993	Court order of June 28, 1993 issued

a. ICPP = Idaho Chemical Processing Plant.

April 1992, DOE decided to phase out reprocessing for material recovery, resulting in the shutdown of the reprocessing operation.

Spent naval nuclear fuel handling at the Naval Reactors Facility originated in 1957 with the construction of the Expended Core Facility. The original building contained a water pit and shielded cells, which are connected to the water pit by transfer tunnels. The Expended Core Facility examines spent nuclear fuel from operating naval ships and from prototype naval reactors. The examinations support research and development for naval fuel quality improvement. Over the years, the Navy made additions and improvements at the Naval Reactors Facility site, including the construction and operation of three prototype reactors and facilities for training naval nuclear powerplant operators. The Naval Nuclear Propulsion Program is placing the prototype reactors, which have reached the ends of their useful lives, in layup. All training is expected to end before DOE issues the Record of Decision for this Environmental Impact Statement (EIS). Expended Core Facility activities are continuing. Appendix D describes the Naval Reactors Facility in more detail.

In 1965 the United States entered into a contract with Public Service Company of Colorado, with which the United States agreed to lease special nuclear material to Public Service Company of

Colorado for fuel at the Fort St. Vrain Nuclear Power Plant. In 1980, the United States and Public Service Company of Colorado modified the 1965 contract, requiring DOE to accept returned Fort St. Vrain spent nuclear fuel at the INEL. From 1980 to 1986, Public Service Company of Colorado made approximately 120 shipments of Fort St. Vrain spent nuclear fuel to the INEL.

In 1974 the National Reactor Testing Station became the Idaho National Engineering Laboratory. The INEL mission broadened to include research and engineering for nonnuclear programs and environmental restoration and waste management activities.

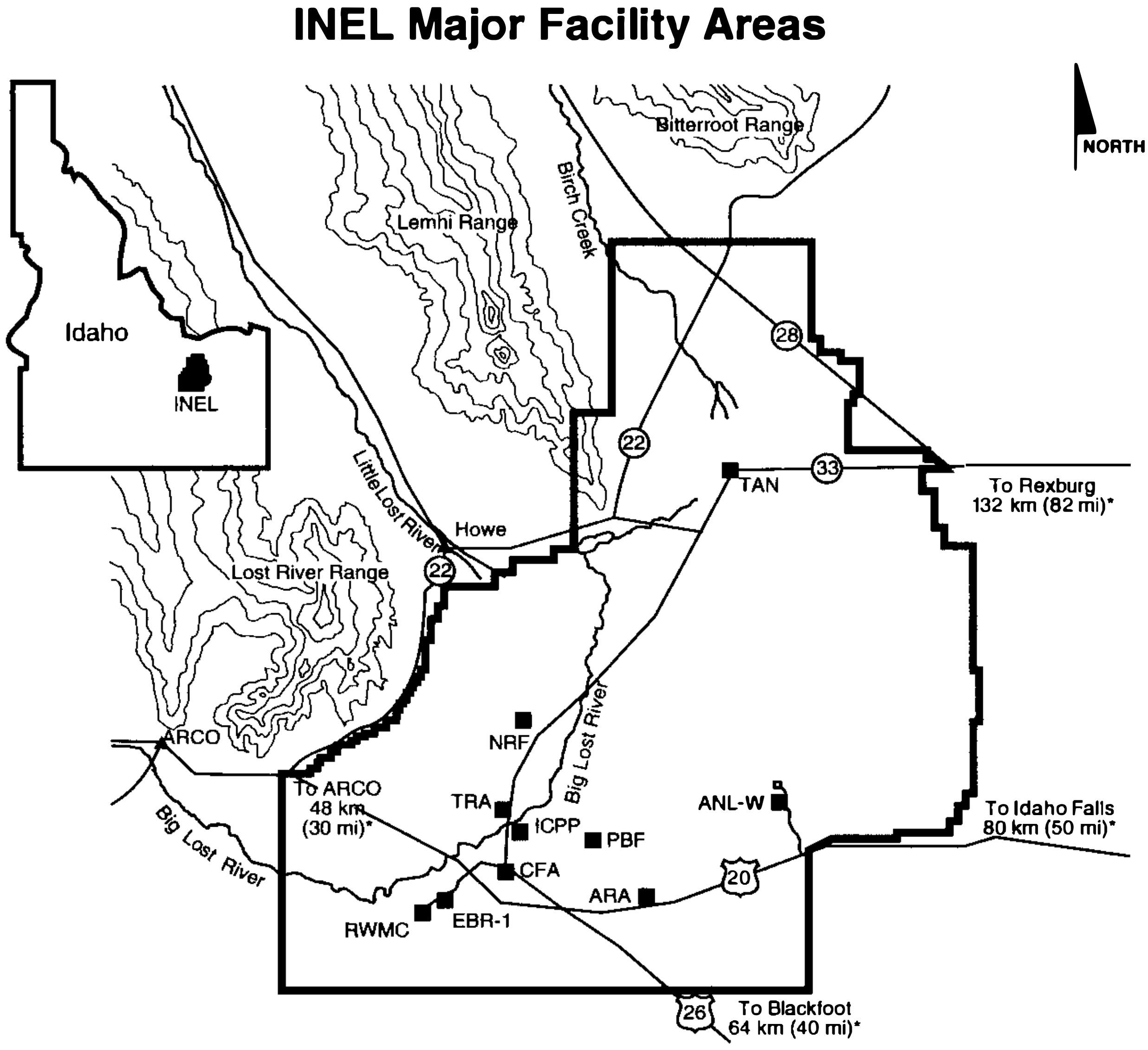
In the early 1980s, pursuant to the West Valley Demonstration Project Act (42 USC 2021a) and a court order, DOE agreed to accept 125 special case commercial reactor spent nuclear fuel assemblies located at the state-owned Western New York Nuclear Service Center. DOE began a project to demonstrate the viability of a transportable spent nuclear fuel storage cask, with the intention of shipping the fuel to the INEL. Based on this, New York State Energy Research and Development Authority, which has jurisdiction over the center, has allowed continued storage until DOE obtained U.S. Nuclear Regulatory Commission Certificates of Compliance, which have been issued. The fuel remains at West Valley awaiting the Record of Decision for this EIS.

In addition to the naval and INEL-generated fuel on the site, some special-case spent nuclear fuel, such as fuel from university reactors, has been shipped directly to the Idaho Chemical Processing Plant for storage. Damaged fuel from the 1979 Three Mile Island accident was shipped directly to Test Area North for examination and storage as part of a research mission.

In 1990, DOE issued an Environmental Assessment and Finding of No Significant Impact for Public Service Company of Colorado shipments of Fort St. Vrain spent nuclear fuel to the INEL. The State of Idaho challenged the adequacy of the Environmental Assessment and, in June 1993, the United States District Court for the District of Idaho found for the State and ordered DOE to prepare this EIS. A DOE appeal of the order resulted in a December 1993 amendment that governs the DOE schedule and obligation for preparing the EIS.

2.1.2 Current Activities at Spent Nuclear Fuel-Related Facilities

Six major facility areas at the INEL (Figure 2-1) store spent nuclear fuel: Argonne National Laboratory - West, Idaho Chemical Processing Plant, Naval Reactors Facility, Power Burst Facility,



*Miles from Central Facilities Area

Legend:

ARA	Auxiliary Reactor Area
ANL-W	Argonne National Laboratory-West
CFA	Central Facilities Area
EBR-1	Experimental Breeder Reactor - I
ICPP	Idaho Chemical Processing Plant
NRF	Naval Reactors Facility
PBF	Power Burst Facility
RWMC	Radioactive Waste Management Complex
TAN	Test Area North
TRA	Test Reactor Area

Miles 0 2 4 6 8
Kilometers 0 4 8 12

PJ20-1

Figure 2-1. Major facility areas located at the Idaho National Engineering Laboratory site.

Test Area North, and Test Reactor Area. Spent fuel at the INEL is kept in a variety of dry and wet configurations. The total amount of spent nuclear fuel at the INEL accounts for about 10 percent (by weight of heavy metal) of the spent nuclear fuel in the DOE complex (DOE 1993).

Table 2-2 lists the primary INEL spent nuclear fuel storage facilities, the types of fuel in storage, and the storage configurations. Figure 2-2 indicates the relative proportion of fuel at these facilities. The number and variety of wet and dry storage configurations currently in use at the INEL is largely the result of the different purposes for the facilities (e.g., at-reactor storage, storage research and development, reprocessing, and fuel research and development). The condition of the spent nuclear fuel in storage is generally good with the notable exception of the fuel in the Underwater Fuel Storage Facility (CPP-603). The following paragraphs briefly describe each primary facility area that manages spent nuclear fuel.

The Argonne National Laboratory - West generates spent nuclear fuel as a result of research and development activities related to advanced reactor design. DOE has brought small quantities of spent nuclear fuel from other reactors to this facility to support these activities. Reactors at Argonne National Laboratory - West are the Experimental Breeder Reactor II, the Transient Reactor Test Facility, the Zero Power Physics Reactor, and the Neutron Radiography Reactor. Storage facilities include both wet (including molten sodium) and dry configurations.

The Idaho Chemical Processing Plant historically received spent nuclear fuel from many onsite and offsite reactors for reprocessing (i.e., the recovery of uranium for reuse). However, DOE decided to phase out reprocessing activities in 1992. The new mission for this facility area is receipt and storage, plus research and development of technologies in support of the disposition of spent nuclear fuel. The Idaho Chemical Processing Plant stores virtually all types of spent nuclear fuel except production reactor fuel [i.e., fuel from Hanford Site and Savannah River Site (SRS) production reactors]. It stores nonproduction aluminum-based spent nuclear fuel. This facility uses both wet and dry storage configurations.

The Naval Reactors Facility includes the Expended Core Facility, which receives and examines naval spent nuclear fuel to support fuel development and performance analyses. In addition, the Expended Core Facility removes structural support material from fuel assemblies before the transfer of the fuel portion to the Idaho Chemical Processing Plant for interim storage.

Table 2-2. Major INEL spent nuclear fuel storage facilities.

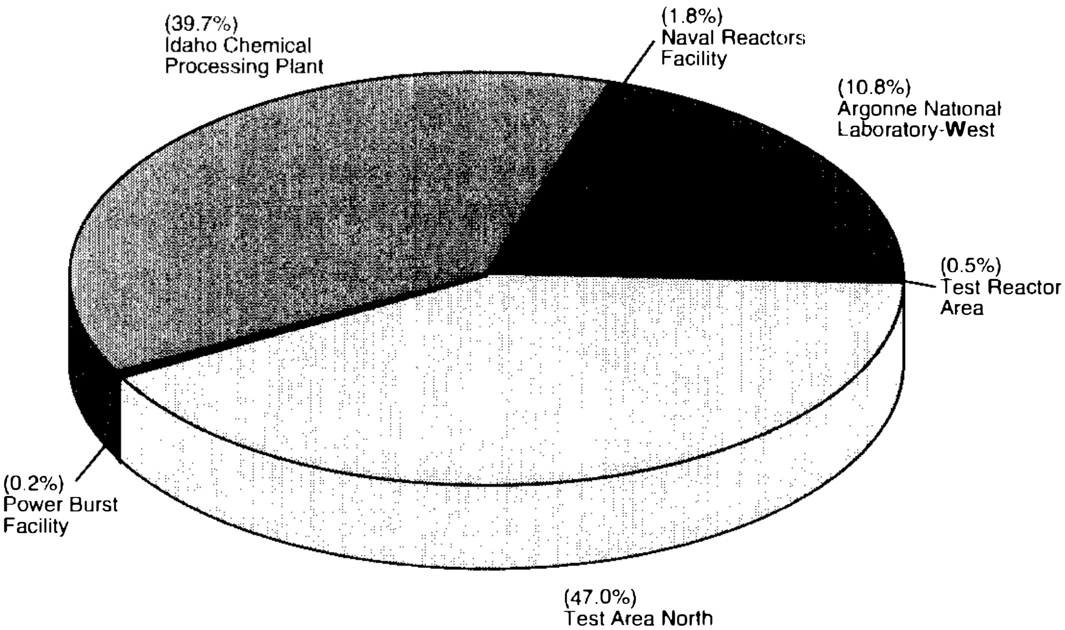
Facility[a]	Storage Type[b]	Fuel Type[c] 1	2	3	4	5	6a	6b	6c
Argonne National Laboratory - West									
Experimental Breeder Reactor II	Liquid sodium						•		
Hot Fuel Examination Facility	Dry						•		
Neutron Radiography Reactor	Wet						•		
Radioactive Scrap and Waste Facility	Dry						•		
Transient Reactor Test Facility	Dry							•	
Idaho Chemical Processing Plant									
Underwater Fuel Storage Facility[d]	Wet	•	•				•	•	
Irradiated Fuel Storage Facility	Dry				•				
Fuel Storage Area/Fluorinel Dissolution Process Cell	Wet	•	•				•	•	
Underground Storage Facility	Dry				•				
Naval Reactors Facility									
Expended Core Facility	Wet	•				•			
Expended Core Facility Rail Siding	Dry	•							
Power Burst Facility									
Power Burst Facility Storage Canal	Wet						•		
Test Reactor Area									
Materials Test Reactor Canal	Wet					•		•	
Advanced Reactivity Measurement Facility	Wet		•						
Coupled Fast Reactivity Measurement Facility	Wet		•						
Advanced Test Reactor Canal	Wet		•						
Test Area North									
Test Area North Pool	Wet					•			
Test Area North Pad	Dry					•			

a. This table lists the major spent fuel storage facilities. Other facilities (e.g., laboratories) might periodically contain small quantities of spent nuclear fuel.

b. Wet storage involves water-filled pools. Dry storage involves a variety of configurations (e.g., casks, wells, buildings).

c. The spent fuel types are as follows:
   1. Naval-type fuel
   2. Savannah River Site production fuels and other aluminum-clad fuels
   3. Hanford Site production fuels
   4. Graphite fuels
   5. Special case commercial fuels
   6a. Experimental reactors - stainless steel-clad fuels
   6b. Experimental reactors - zirconium-clad fuels
   6c. Experimental reactors - other fuel configurations

d. Spent nuclear fuel storage at this facility will cease by December 31, 2000, as part of an agreement between DOE and the State of Idaho.



Note: Percentages represent metric tons of heavy metal of spent nuclear fuel

PJ20-2

Figure 2-2. Existing (1995) distribution of INEL SNF.

The Power Burst Facility reactor was placed in operational standby in 1992. A limited amount of spent nuclear fuel from this facility remains in wet storage, in a storage pool that is in good condition, but it is small and uneconomical to use. DOE plans to remove the fuel from this facility by 1996.

DOE has used Test Area North for commercial reactor fuel research. The large Test Area North Hot Shop and Hot Cells have supported the Loss of Fluid Test and commercial nuclear fuel testing, including dry cask storage demonstration. Test Area North stores special case commercial fuel (including Three Mile Island Unit 2 core debris) and DOE experimental fuel similar to commercial nuclear fuel.

Test Reactor Area has historically operated a number of test reactors, but the Advanced Test Reactor and its associated Critical Facility are the only reactors now operating. Most spent nuclear fuel at this area is associated with the Test Reactor Area reactors, which utilized aluminum-based fuels. In addition, DOE stores small amounts of special case commercial, foreign, and Power Burst Facility spent nuclear fuel at Test Reactor Area in the Materials Test Reactor basin. All spent nuclear fuel in storage at the Test Reactor Area is in water-filled pools (DOE 1993).

2.1.3 Spent Nuclear Fuel Mission

The INEL spent nuclear fuel mission is to manage DOE-owned spent fuel cost-effectively and in a way that protects the safety of INEL workers, the public, and the environment. As the lead laboratory for the DOE Spent Nuclear Fuel Program, the INEL provides support to the Office of Spent Fuel Management and coordinates the development of an integrated program for DOE.

The main focus of near-term activities is the accurate quantification and characterization of DOE-owned spent nuclear fuel, identification of spent nuclear fuel management facilities and their conditions, identification of safe interim storage for existing and new spent nuclear fuel, and identification of technologies and requirements to place DOE spent nuclear fuel in safe interim storage. Long-term activities include the development of final waste acceptance criteria requirements and stabilization technologies for alternate fuel disposition, construction of facilities to stabilize fuel to meet waste disposal requirements, processing of the fuel to a final waste form, and transportation of the waste form for disposition.

2.2 Regulatory Framework for Spent Nuclear Fuel Management

This section summarizes State of Idaho laws and regulations that apply to spent nuclear fuel management at the INEL. Volume 1, Section 7.2, provides summary information for Federal laws and regulations, Executive Orders, and DOE Orders. Volume 2, Chapter 2, provides information on National Environmental Policy Act reviews related to site-specific decisions that have potential environmental impacts. Volume 2, Chapter 7, provides information on regulatory permits that the INEL holds or for which it has applied.

The Idaho Environmental Protection and Health Act (Idaho Code, Title 39, Chapter 101 et seq.) establishes general provisions for the protection of the environment and public health. The Act created the Idaho Department of Health and Welfare and its Division of Environmental Quality, thereby consolidating all state public health and environmental protection activities in one department. The Act authorizes the Department to promulgate standards, rules, and regulations related to water and air quality, noise reduction, and solid waste disposal; and grants authority to issue required permits, collect fees, establish compliance schedules, and review plans for the construction of sewage and public water treatment and disposal facilities.

The Idaho Water Pollution Control Act (Idaho Code, Title 39, Chapter 36) authorizes the Department of Health and Welfare to protect the waters of Idaho. This law contains general language on the prevention of water pollution and the provision of financial assistance to municipalities.

The Idaho Department of Health and Welfare is also responsible for the enforcement and implementation of the Hazardous Waste Management Act of 1983, as amended (Idaho Code, Title 39, Chapter 44), which provides for the protection of health and the environment from the effects of improper or unsafe management of hazardous wastes and for the establishment of a tracking or manifesting system for these wastes. This program is intended to be consistent with, and not more stringent than, the Federal regulations established under the Resource Conservation and Recovery Act (RCRA). At this time, Idaho has primacy over hazardous and mixed waste regulations promulgated through July 1, 1990, by the U.S. Environmental Protection Agency. The Hazardous Waste Management Act sets forth requirements for the development of plans that address the identification of hazardous wastes; unauthorized treatment, storage, release, use, or disposal of these wastes; and permit requirements for hazardous waste facilities. Under the authority of this Act, the Idaho Department of

Health and Welfare has promulgated rules and regulations on the transportation, monitoring, reporting, and record keeping of hazardous wastes.

Several INEL facilities have air quality permits from the State, and operate in compliance with permit conditions. Permit applications are currently pending with the State for proposed new or modified emission sources. In April 1991 DOE submitted an inventory of all potential INEL radioactive and criteria pollutant emission sources to the State. The inventory contains the information necessary for the State to issue the INEL a Permit to Operate.

The Idaho Department of Health and Welfare, Division of Environmental Quality, Air Quality Bureau, conducts annual inspections of the INEL to determine if the operating portions of the site are in compliance with the *Rules for the Control of Air Pollution in Idaho*. The most recent inspections were in January 1994. In addition, pursuant to 40 CFR Part 61.94(H), DOE submits to the State an annual report documenting compliance with National Emission Standards for Hazardous Air Pollutants at the INEL.

2.3 Spent Nuclear Fuel Management Program at the INEL

In 1992 the Secretary of Energy directed the Assistant Secretary for Environmental Restoration and Waste Management to develop an integrated, long-term spent nuclear fuel management program. In response to this request, DOE created the Office of Spent Fuel Management (EM-37). This office, which has strategic programmatic responsibilities, has designated the INEL as the program support organization for the DOE Spent Nuclear Fuel Program. In this role, the INEL provides technical support to the Office of Spent Fuel Management and develops site communication and integration for the national program.

As identified in the *Spent Fuel Working Group Report on Storage of the Department's Spent Nuclear Fuel and Other Reactor Irradiated Nuclear Materials and Their Environmental, Safety and Health Vulnerabilities*, Volume I (DOE 1993), some of the current storage facilities at the INEL are inadequate for extended interim storage, and additional storage facilities or modifications might be necessary. In February 1994, DOE issued, *Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities, Phase I* (DOE 1994a), followed by a Phase II Plan in April 1994 (DOE 1994b) and a Phase III Plan in October 1994 (DOE 1994c), which identified specific corrective actions to address the spent nuclear fuel vulnerabilities. At the INEL, many of the corrective actions have been completed or are currently underway. The spent nuclear fuel storage pools at Test Area North, Power Burst Facility, and the Underwater Fuel Storage Facility do not comply with new facility regulatory requirements. The INEL plans to move spent nuclear fuel from the CPP-603 Underwater Fuel Storage Facility by December 31, 2000. To stabilize this fuel for storage, the INEL also plans to install canning equipment in the Irradiated Fuel Storage Facility hot cell. This equipment is scheduled for operation by late 1995. To the extent of its existing capability, DOE could consolidate spent nuclear fuel at the Power Burst Facility, the Idaho Chemical Processing Plant, and the Test Area North at the Idaho Chemical Processing Plant as a result of implementing the management alternatives described in Chapter 3. These activities and other planned actions for which National Environmental Policy Act review will be completed before the Record of Decision of this EIS were analyzed under the No-Action Alternative (see Chapter 3).

Each of the specific INEL spent nuclear fuel Plan of Action projects could result in emissions, worker exposures, and other potential environmental impacts. The potential environmental impacts that could result from each project or corrective action item were not analyzed individually but were collectively enveloped by the spent nuclear fuel management activities reported and analyzed for each alternative. Successful completion of the corrective actions would significantly reduce the near-term environmental, safety, and health risks associated with spent fuel storage at INEL.

The INEL has provided support in the development of dry at-reactor storage of special case commercial spent nuclear fuel in accordance with the requirements of the Nuclear Waste Policy Act of 1982 and its 1987 amendments. Dry-storage demonstrations and research at the INEL contributed to the granting of NRC licenses to several utilities for the construction and operation of dry-storage facilities at reactor sites. Research at these facilities is demonstrating the technical feasibility and the economics of adding dry storage capacity in metal or concrete spent fuel storage casks at reactor sites.

3. SPENT NUCLEAR FUEL MANAGEMENT ALTERNATIVES

Chapter 3 describes the alternatives for spent nuclear fuel management as they relate to the Idaho National Engineering Laboratory (INEL) and summarizes and compares potential environmental consequences for each alternative. Chapter 5 contains full descriptions of the consequences of implementing the alternatives.

3.1 Description of Alternatives

DOE has identified five spent nuclear fuel management alternatives:

Alternative 1 - No Action
Alternative 2 - Decentralization (2a, 2b, and 2c)
Alternative 3 - 1992/1993 Planning Basis
Alternative 4 - Regionalization (4a and 4b)
Alternative 5 - Centralization (5a and 5b)

Table 3-1 summarizes the actions that would result from the implementation of these alternatives at the INEL. For each alternative, this table summarizes the proposed transportation, stabilization, storage, research and development, and naval-type fuel examination activities. For alternatives 2, 4, and 5, it identifies a number of options.

The analysis of each alternative considers, as appropriate, existing and projected spent nuclear fuel inventories, existing spent nuclear fuel wet and dry storage facilities, the construction of storage facilities and associated stabilization facilities to achieve interim management objectives, and the relocation of the spent nuclear fuel as appropriate to proposed interim storage facilities.

Table 2-2 lists existing spent nuclear fuel storage facilities with associated type(s) of storage and fuel. Table 3-2 lists the potential facilities and projects required for specific alternatives. DOE has based the potential environmental consequences for each alternative on the existing and proposed facilities and projects listed in Tables 2-2 and 3-2, respectively.

Table 3-1. Summary of spent nuclear fuel management alternatives at the Idaho National Engineering Laboratory.[a]

	Alternative	Description	Transportation	Stabilization	Storage	Research and Development	Naval-Type Fuel Examination
1.	No Action	Minimum actions necessary for continued safe/secure management of SNF.	No shipment to or from the INEL after transition period. Onsite transport of SNF limited to that required for safe storage. Receipt of naval-type SNF during transition period.	Limited to those minimum actions required to store SNF safely.	Minimum facility upgrade/replacement to support safe storage. Replacement dry storage facility for Test Area North storage pool.	Existing R&D activities for SNF management would continue.	Shipment to INEL and examinations after a transition period would be phased out.
2.	Decentralization	SNF would be stored close to existing locations with limited shipments to DOE facilities.	Same as Alternative 1 plus: · Receipt of non-DOE domestic and foreign research SNF · Receipt of naval-type fuels for examination and reshipment (option 2c) · Onsite SNF transfer for consolidation	Same as Alternative 1	Same as Alternative 1	Treatment technology and R&D activities for DOE SNF management and disposal permitted.	Three options: · Options 2a and 2b are the same as for Alternative 1 · Option 2c would enable the continued receipt of naval-type fuels for inspection at the ECF and a return to originating shipyards. The ECF Dry Cell Construction project would be completed.
3.	1992/1993 Planning Basis	DOE 1992, 1993 planning basis for DOE and naval-type SNF management.	· Receipt of some foreign, Fort St. Vrain, West Valley, and non-DOE domestic research SNF. · Onsite transfer. · Receipt of naval-type SNF for examination at the ECF and transfer to the ICPP for interim storage.	Stabilization as planned: new canning and characterization facility required.	Replacement dry storage facility for Test Area North storage pool. New dry fuel storage facility and increased rack capacity in storage pools.	Same as Alternative 2 plus: Electrometallurgical Process Demonstration Project at ANL-W	ECF continues operation as planned. The ECF Dry Cell Construction would be completed.

Table 3-1. (continued).

Alternative	Description	Transportation	Stabilization	Storage	Research and Development	Naval Type Fuel Examination
4a. Regionalization by Fuel Type	Existing and new SNF redistribution based on similarity of fuel type. All SNF in DOE complex would be managed at Hanford Site, INEL, or Savannah River Site.	Distribute existing and projected SNF to the INEL based primarily on fuel type.	SNF to be retained at the INEL would be stabilized as planned; for SNF to be shipped to regional sites, any stabilization beyond that required for transportation would be performed at the regional site.	Same as Alternative 3	Same as Alternative 3	Same as Alternative 3
4b(1). Regionalization by Geography (INEL)	Existing and projected Western DOE and naval-type SNF would be managed at the INEL.	Shipment of all Western SNF in DOE complex to the INEL.	Sites shipping SNF to INEL would stabilize for purpose of transportation; any further stabilization would be performed at the INEL.	Construction of new facilities for SNF storage.	Same as Alternative 3	Same as Alternative 3
4b(2). Regionalization by Geography (Elsewhere)	Existing and projected Western DOE and naval-type SNF would be managed at Hanford Site or Nevada Test Site.	Existing INEL SNF shipped offsite to selected Western Regionalization site.	SNF at the INEL would be stabilized at a canning, characterization, and shipping facility prior to shipment offsite; other SNF would be stabilized as required at the selected Regionalization site.	Phaseout of all SNF storage facilities.	Phaseout of all R&D activities at the INEL except the Electrometallurgical Process Demonstration Project at ANL-W.	Same as Alternative 1
5a. Centralization at Other DOE Sites	Existing and projected DOE and naval-type SNF would be managed at Hanford Site, Savannah River Site, Oak Ridge, or Nevada Test Site.	Existing INEL SNF shipped offsite to selected centralization site.	SNF at the INEL would be stabilized at a canning, characterization, and shipping facility prior to shipment offsite; other SNF would be stabilized as required at the selected Centralization site.	Phaseout of all SNF storage facilities.	Phaseout of all R&D activities at the INEL except the Electrometallurgical Process Demonstration Project at ANL-W.	Same as Alternative 1

Table 3-1. (continued).

	Alternative	Description	Transportation	Stabilization	Storage	Research and Development	Naval Type Fuel Examination
5b.	Centralization at the INEL	Existing and projected DOE and naval-type SNF would be managed at the INEL.	Shipment of all SNF in DOE complex to the INEL.	Sites shipping SNF to INEL would stabilize for purpose of transportation; any further stabilization would be performed at the INEL.	Construction of new facilities for SNF storage.	Same as Alternative 3	Same as Alternative 3

a. ANL-W = Argonne National Laboratories - West; DOE= U.S. Department of Energy; ECF= Expended Core Facility; ICPP = Idaho Chemical Processing Plant; INEL = Idaho National Engineering Laboratory; R&D = research and development; SNF = spent nuclear fuel.

Table 3-2. Potential spent nuclear fuel projects required for each alternative[a].

Facility/Project Name	Alternatives					
	1. No Action	2. Decentralization	3. 1992/1993 Planning Basis	4.[b] Regionalization	5a. Centralization at Other DOE Sites	5b. Centralization at the INEL
Test Area North Pool Fuel Transfer	•	•	•	•	•	•
Increased Rack Capacity for CPP-666			•	•		•
Additional Increased Rack Capacity (CPP-666)			•	•		•
Dry Fuels Storage Facility			•	•	•[c]	•[d]
EBR-II Blanket Treatment			•	•		•
Expended Core Facility Dry Cell Construction		•[e]	•	•		•
Fort St. Vrain Spent Fuel Shipment and Storage			•	•		•
Spent Fuel Processing						•
Electrometallurgical Process Demonstration Project at ANL-W FCF[f]			•	•	•	•

a. Appendix C of Volume 2 contains detailed descriptions of the spent nuclear fuel projects identified in this table.
b. Project actions listed are for option 4a only. For purpose of analysis, option 4b(1) is the same as Alternative 5b. Option 4b(2) is the same as Alternative 5a.
c. Includes canning, characterization, and shipping only.
d. Expanded scope.
e. The Expended Core Facility Dry Cell Construction under Alternative 2 would occur for option 2c only.
f. Argonne National Laboratories-West Fuel Cycle Facility.

The alternatives involving the interim storage of naval spent nuclear fuel at sites other than the INEL include a transition period, which would start on June 1, 1995, and continue for approximately 3 years. During this period, approximately 80 shipments of naval spent nuclear fuel would occur to the Expended Core Facility for examination and subsequent shipment to the Idaho Chemical Processing Plant for storage. After this transition period, DOE would phase out the Expended Core Facility such that the worker total at the facility would decline to about 10 by 2001. Appendix D describes this transition period.

3.1.1 Alternative 1: No Action

Table 3-1 lists the basic actions expected under this alternative. This alternative would be restricted to the minimum actions necessary for the continued safe and secure management of spent nuclear fuel. Table 3-3 lists the existing inventory of spent nuclear fuel at the INEL. This alternative is not a status quo condition in terms of spent nuclear fuel receipts (unlike Alternative 3, under which operations would continue in accordance with the 1992/1993 planning basis). Rather, DOE would maintain spent nuclear fuel close to defueling or current storage locations with minimal facility upgrades or replacements.

DOE would continue the operation of the following existing spent nuclear fuel-related facilities: the Fuel Storage Area/Fluorinel Dissolution Process Cell; CPP-603 Underwater Fuel Storage Facility (until 2000); Irradiated Fuel Storage Facility; Underground Storage Facility; Power Burst Facility storage canal; Advanced Test Reactor canal; Advanced Reactivity Measurement Facility; Coupled Fast Reactivity Measurement Facility; Materials Test Reactor canal; Test Area North Pool and Test Pad; Argonne National Laboratory - West Hot Fuel Examination Facility, Radioactive Scrap and Waste Facility, Transient Reactor Test Facility, Zero Power Physics Reactor, and Neutron Radiography Reactor pool. Table 2-2 lists the type(s) of storage and spent nuclear fuels associated with each facility.

3.1.1.1 Transportation. Under this alternative, the INEL would neither receive nor ship spent nuclear fuel except for naval spent fuel during a transition period. DOE would continue to transfer the Advanced Test Reactor canal spent nuclear fuel to the Idaho Chemical Processing Plant. In addition, DOE could transfer other spent nuclear fuel at the INEL site (e.g., Test Reactor Area, Test Area North Pad, Power Burst Facility storage canal, Experimental Breeder Reactor-II, and Naval Nuclear

Table 3-3. Spent nuclear fuel inventory for each alternative by 2035 (metric tons of heavy metal).[a,b,c]

Fuel Type	1. No Action[d]	2. Decentralization	3. 1992/1993 Planning Basis	4a. Regionalization by Fuel Type	4b(1)[e] Regionalization by Geography (INEL)	5a. Centralization at Other DOE Sites	5b Centralization at the INEL
Naval-type	10.23	N/C[f]	+55.00	+55.00	+55.00	-10.23	+55.00
Aluminum-clad	2.91	11.02	+12.09	-2.91	+5.85	-2.91	+210.18
Hanford	None	None	None	None	+2,103.17	None	+2,103.17
Graphite	11.60	N/C	+16.00	+16.01	+16.01	-11.60	+16.01
Special case commercial	122.88	+0.03	+26.69	+33.63	+2.30	-122.88	33.63
Stainless-steel-clad	77.43	+1.08	+1.19	+19.08	+12.69	-77.43	+19.08
Zircaloy-clad	49.09	+0.67	+0.670	+28.90	+15.75	-49.09	+28.90
Other	0.01	+0.82	+0.82	+1.69	+0.28	-0.01	+1.69
Net increase (+)/ decrease (-)	-	+13.62	+112.47	+151.41	+2,211.05	-274.14	+2,467.66
TOTAL	274.14	287.76	386.61	425.55	2,485.19	0	2,741.80

a. Source: Wichmann (1995).
b. To convert metric tons to tons, multiply by 1.10. Heavy metals are uranium, plutonium, and thorium.
c. The values may not sum exactly due to rounding.
d. The No-Action Alternative represents the present inventory and projections and serves as the basis for determining the net increase or decrease for each type of spent nuclear fuel for each of the other alternatives.
e. Regionalization 4b(2), Regionalization by Geography (Elsewhere), assumes all spent nuclear fuel inventories at the INEL go to the Nevada Test Site or Hanford Site. Inventories for 4b(2) would equal those listed for Alternative 5a.
f. N/C = No change from the No-Action Alternative.

Propulsion Program prototype reactors at the Naval Reactors Facility) to the Idaho Chemical Processing Plant to the extent of its storage capability.

3.1.1.2 Stabilization. Due to the deteriorated condition of some of the fuel in the CPP-603 Underwater Fuel Storage Facility, additional canning and characterization capabilities would be necessary to stabilize this fuel for safe transport and subsequent storage. DOE has scheduled the installation and operation of new fuel canning and characterization equipment in the Irradiated Fuel Storage Facility, which could provide these capabilities, by late 1995. (The installation of such equipment would be a minor upgrade and would have a smaller extent than similar actions described under Alternatives 3, 4, and 5.) DOE could perform other required stabilization of spent nuclear fuel at the INEL in either the Remote Analytical Laboratory or the Fluorinel Dissolution Process Hot Cell.

3.1.1.3 Storage. DOE has identified the CPP-603 Underwater Fuel Storage Facility as one of five complex-wide spent nuclear fuel storage facilities that exhibit the greatest vulnerabilities according to selected criteria and, therefore, has selected this facility for priority attention (DOE 1993b). As part of the August 9, 1993, agreement between the Secretaries of the Department of Energy and the Department of the Navy and the Governor of Idaho to phase out storage operations in the 45-year old CPP-603 facility, one goal of this and the other alternatives would be to remove spent nuclear fuel from underwater storage in the North and Middle Basins of the CPP-603 facility by the end of 1996 and from the South Basin of this facility by the end of 2000 (DOE 1993a). DOE would relocate this material to the Fuel Storage Area at the Idaho Chemical Processing Plant.

At the Argonne National Laboratory-West, the spent nuclear fuel stored at the Hot Fuel Examination Facility and the Radioactive Scrap and Waste Facility, primarily Experimental Breeder Reactor-II fuel and blanket elements, would remain in dry storage until its potential processing in the Fuel Cycle Facility. At the Experimental Breeder Reactor-II site, DOE would use dry storage with the exception of the Neutron Radiography Reactor pool fuel. The Test Area North Pool Fuel Transfer project would continue, resulting in the relocation of Test Area North spent pool contents into dry cask storage at the Idaho Chemical Processing Plant by 1998. The dry cask storage required for this project is not related to the Dry Fuels Storage Facility.

DOE would start no new projects to increase spent nuclear fuel storage capacity because there is sufficient storage capacity to meet No-Action storage needs. The planning of spent nuclear fuel storage projects such as the Dry Fuels Storage Facility and Additional Increased Rack Capacity for the Fuel Storage Area would stop.

3.1.1.4 Research and Development. There would be only limited spent nuclear fuel research and development. Existing spent nuclear fuel management research and development projects would continue. Existing facilities such as the Process Improvement Facility, the Remote Analytical Laboratory, and the Pilot Plant Facility would support continuing research and development work.

3.1.1.5 Naval-Type Fuel Examination. After a transition period, DOE would cease shipments of naval spent nuclear fuel to the INEL and would phase out the Expended Core Facility. DOE would make onsite shipments of the "library fuel" (a representative sampling of different fuel types maintained for reference purposes) and the spent nuclear fuel that originated at the prototype sites at the Naval Reactors Facility to the Idaho Chemical Processing Plant.

3.1.2 Alternative 2: Decentralization

Under this alternative, DOE could transport fuel for safety or research and development activities. In addition, DOE could undertake actions for safety it deemed desirable, though not essential, and could perform spent nuclear fuel treatment and research and development. As listed in Table 3-3, the anticipated spent nuclear fuel inventory for this alternative would be slightly greater than the inventory for Alternative 1, with the increase consisting primarily of aluminum-clad and stainless-steel-clad spent nuclear fuel from university and foreign research and experimental reactors.

3.1.2.1 Transportation. This alternative assumes that the INEL would accept primarily limited shipments of spent nuclear fuel from offsite sources into the Fuel Storage Area (e.g., DOE or university reactors) after the Record of Decision for this EIS (1995). Onsite transfers could occur from the Fuel Storage Area to the Storage Facility or the Irradiated Fuel Storage Facility. DOE would consolidate the spent nuclear fuel in the Advanced Test Reactor and in the Materials Test Reactor and Power Burst Facility canals at the Idaho Chemical Processing Plant for canning, characterization, and storage.

As in the No-Action Alternative, there would be a transition period during which the Naval Nuclear Propulsion Program would ship naval spent nuclear fuels to the Expended Core Facility for examination and subsequent shipment to the Idaho Chemical Processing Plant for storage.
Section 3.1.2.5 describes the transportation of naval spent fuels that would occur after the transition period.

3.1.2.2 Stabilization. DOE would use the canning and characterization equipment identified in Section 3.1.1.2 to stabilize spent nuclear fuel removed from the CPP-603 Underwater Fuel Storage Facility for interim underwater storage.

3.1.2.3 Storage. As in Alternative 1, DOE would transfer the spent nuclear fuel in the CPP-603 Underwater Fuel Storage Facility to the Fuel Storage Area by 2000. DOE would continue to use the Underground Storage Facility and the Irradiated Fuel Storage Facility for existing spent nuclear fuel inventory and transfers of other spent nuclear fuel based on safety analyses. DOE would upgrade or increase fuel storage capacity at the INEL as required.

The Test Area North Pool Fuel Transfer project would result in the relocation of the contents of Test Area North spent nuclear fuel into dry storage at a pad at the Idaho Chemical Processing Plant.

3.1.2.4 Research and Development. The development of technology for the disposition of spent nuclear fuel would continue. Research and development activities would include laboratory and pilot plant testing, continued repository performance assessments and waste acceptance criteria development, and the characterization of spent nuclear fuel. Shipments of samples or selected spent nuclear fuel assemblies to offsite DOE facilities would be necessary.

3.1.2.5 Naval-Type Fuel Examination. DOE would consider three options for naval reactor spent nuclear fuel receipt and shipment. Under options 2a and 2b, DOE would stop shipments of naval spent nuclear fuel to the INEL and would shut down the Expended Core Facility. Option 2c would enable the continued receipt of naval-type fuel for examination at the Expended Core Facility and its return to the originating shipyards for storage in transport casks. Chapter 3 of Appendix D further describes these options. As with Alternative 1, each option would require approximately a 3-year transition period. During this period, DOE would transport spent nuclear fuel in shipping containers to the Expended Core Facility, unload the containers, and use them to support additional refuelings and defueling.

3.1.3 Alternative 3: 1992/1993 Planning Basis

This alternative is consistent with DOE plans at the INEL before the injunction that stopped spent nuclear fuel shipment to the INEL; it assumes a 40-year planning horizon for the continued transportation, receipt, stabilization, and storage of spent nuclear fuel. As with Alternative 1, DOE would continue the maintenance and operation of existing spent nuclear fuel-related facilities; however, some consolidation of INEL facilities could occur. DOE would send newly generated spent nuclear fuel to either the INEL or the Savannah River Site. DOE would assess the construction of new facilities to accommodate current and projected spent nuclear fuel management requirements.

The amount of spent nuclear fuel at the INEL under this alternative would be greater than that for either Alternative 1 or 2 (see Table 3-3) because this alternative assumes that the INEL would manage, before stabilization and disposal, its present inventory (see Alternative 1) plus additional receipts of DOE spent nuclear fuel, including the following:

- Naval-type spent nuclear fuel
- Approximately half of the aluminum-clad spent nuclear fuel from university and foreign research and experimental reactors
- All Training Reactor Isotopics General Atomics (TRIGA) spent nuclear fuels from the Hanford Site and approximately half of that from foreign, DOE, and university reactors
- Fort St. Vrain spent nuclear fuel from Public Service of Colorado
- Special case commercial pressurized water reactor and boiling water reactor spent nuclear fuel from the DOE facility in West Valley, New York
- Miscellaneous spent nuclear fuel types from such DOE sites as Los Alamos, New Mexico, and Oak Ridge, Tennessee, and from university reactors and other locations

3.1.3.1 Transportation. DOE would consolidate the spent nuclear fuel in the Test Reactor Area (Advanced Test Reactor canal, Materials Test Reactor canal, and Coupled Fast Reactivity Measurements Facility and Advanced Reactivity Measurement Facility canal) and the Power Burst Facility at the Idaho Chemical Processing Plant for canning and dry storage.

The INEL would receive and temporarily store new spent nuclear fuels in the Fuel Storage Area. Transfers could occur from the Fuel Storage Area to the Underground Storage Facility or the Irradiated Fuel Storage Facility or, when available, the dry storage vaults at the proposed Dry Fuels Storage Facility.

At present, DOE is transferring spent nuclear fuel from the Advanced Test Reactor Canal to the Idaho Chemical Processing Plant. DOE would maintain this canal for the storage and management of its recyclable fuel assemblies until the reactor no longer had a mission. The Experimental Breeder Reactor-II spent nuclear fuel in storage would remain at Argonne National Laboratory-West. As with Alternative 2, the Test Area North Pool Fuel Transfer project would result in the relocation of the contents of the Test Area North spent nuclear fuel pool to dry storage at a pad at the Idaho Chemical Processing Plant.

3.1.3.2 Stabilization. DOE would complete a new Canning and Characterization Facility with appropriate inspection, stabilization, and packaging equipment to stabilize new receipts of spent nuclear fuel and to prepare fuel currently in underwater storage for dry storage. This facility would be an integral part of the Dry Fuels Storage Facility that DOE would complete under this alternative. Until the Dry Fuels Storage Facility is in service, DOE would use the canning and characterization equipment described under Alternative 1 to stabilize spent nuclear fuel removed from the CPP-603 Underwater Fuel Storage Facility for interim underwater storage.

3.1.3.3 Storage. As with Alternative 2, DOE would upgrade or increase dry fuel storage capacity at the INEL as required. DOE would complete the Fuel Storage Area increased Rack Capacity project in 1997. Coupled with stringent fuel management and, if necessary, temporary storage of some aluminum fuel in stainless steel racks, this project would allow the Fuel Storage Area to accept all of the project spent nuclear fuel receipts until the Additional Increased Rack Capacity project would be completed in 2001. The Additional Increased Rack Capacity project would allow the Fuel Storage Area to accept the projected spent nuclear fuel receipts until the Dry Fuels Storage Facility project would become available in 2005. The INEL would receive the Fort St. Vrain spent nuclear fuel in the Irradiated Fuel Storage Facility on a space-available basis or in the new vault storage in the Dry Fuels Storage Facility. Modifications to the Irradiated Fuel Storage Facility cask handling equipment would be necessary to accept the new Fort St. Vrain shipping casks.

DOE would continue to use the Underground Storage Facility and the Irradiated Fuel Storage Facility for current inventory and for transfers of other fuel inventories based on safety analyses. Based on these safety analyses, upgrades would be limited to those required for facility safety improvements and for making transfers safely.

3.1.3.4 Research and Development. Spent nuclear fuel research and development would continue as planned, with the construction of a Technology Development Facility. The Electrometallurgical Process Demonstration Project at Argonne National Laboratory - West Fuel Cycle Facility would continue. In addition, Argonne National Laboratory would implement the EBR-II Blanket Processing project under this alternative. The Dry Fuels Storage Facility would develop and demonstrate technology for the dry storage of selected DOE highly enriched uranium fuels.

3.1.3.5 Naval-Type Fuel Examination. The practice of transporting spent nuclear fuel from naval reactors to the Expended Core Facility at the INEL would resume. After an examination, DOE would transfer such fuel to the Idaho Chemical Processing Plant for interim storage pending final disposition. Under this alternative, the Naval Nuclear Propulsion Program would complete the Expended Core Facility Dry Cell Construction project.

3.1.4 Alternative 4: Regionalization

This alternative assumes that DOE would base the spent nuclear fuels shipped between DOE sites and the receipt of fuels from other locations primarily on either geography or fuel type. Alternative 4 offers two options for the redistribution of existing and new spent nuclear fuel:

- Option 4a assumes that DOE would base the spent nuclear fuels shipped between DOE sites and the receipt of fuels from other locations at the INEL, Hanford Site, or the Savannah River Site primarily on fuel type.

- Option 4b assumes that DOE would base the spent nuclear fuels shipped between DOE sites and the receipt of fuels on geography. There would be a single western site at either the Hanford Site, INEL or Nevada Test Site. Option 4b(1) in which the INEL is the western regional site is essentially the same as Alternative 5b. Option 4b(2) in which INEL ships all SNF to another western regional site is the same as Alternative 5a.

3.1.4.1 Transportation. Under option 4a, the INEL would receive all Zircaloy- and stainless-steel-clad spent nuclear fuel. This redistribution would optimize DOE spent nuclear fuel management.

The spent nuclear fuel inventory involved under option 4a would be greater than those for Alternative 1, 2, or 3 because this alternative assumes that the INEL would manage its present inventory plus the following additional spent nuclear fuels (see Table 3-3) prior to stabilization and disposal:

- Naval-type spent nuclear fuel

- All spent nuclear fuel except aluminum-clad fuel and Hanford spent nuclear fuel

- All Training Reactor Isotopics General Atomics spent nuclear fuels from the Hanford Site

- Fort St. Vrain spent nuclear fuel from Public Service of Colorado

- Special case commercial pressurized water reactor and boiling water reactor spent nuclear fuel from the DOE facility in West Valley, New York

Under option 4b(1), DOE would regionalize all western DOE SNF at the INEL. DOE would transport all spent nuclear fuel at other western sites to the INEL. Because the fuel inventory for this alternative would be within 15 percent of that for Alternative 5b, analyses for this option conservatively assume that environmental impacts would be the same as those for as Alternative 5b - Centralization at INEL.

Under option 4b(2), DOE would regionalize all western DOE SNF at either the Nevada Test Site or Hanford Site. DOE would transport spent nuclear fuel at the INEL to the selected western site. As such, this option would be the same as Alternative 5a - Centralization at Other DOE Sites.

3.1.4.2 Stabilization. DOE would stabilize the spent nuclear fuels it would retain at the INEL as planned for Alternative 3, with the construction of such new facilities as a canning and characterization facility and the Dry Fuels Storage Facility. Options 4a and 4b(1) would require such a facility for the receipt and storage of spent nuclear fuel, while option 4b(2) would require stabilization capabilities for shipping spent nuclear fuel. For spent nuclear fuel that the INEL would ship to other regional sites, the receiving site would perform any stabilization beyond that required for transportation.

3.1.4.3 Storage. Under option 4a, DOE would increase dry storage capacity and undertake facility upgrades similar to those described for Alternative 3, with replacements and additions as appropriate. Under option 4b(1), DOE would increase dry storage capacity and undertake facility upgrades similar to those described for Alternative 5b, with replacements and additions as appropriate. Option 4b(2) would not require increased storage capacity and, therefore, there would be no facility upgrades.

3.1.4.4 Research and Development. As with Alternative 3, this alternative would include the continuation of activities related to the treatment of spent nuclear fuel, including research and development (e.g., Electrometallurgical Process Demonstration Project), and the construction of the Dry Fuels Storage Facility. DOE would initiate pilot programs as needed to support future decisions on spent nuclear fuel management and disposition. DOE would use historic data on spent nuclear fuel to provide the bounding case for a determination of the impacts associated with potential pilot program activities.

3.1.4.5 Naval-Type Fuel Examination. Under options 4a and 4b(1), the transportation of spent nuclear fuel from naval reactors to the Expended Core Facility at the INEL would resume. As with Alternative 1, under option 4b(2) DOE would phase out shipments of naval-type spent nuclear fuel to the INEL and would phase out the Expended Core Facility.

3.1.5 Alternative 5: Centralization

Under this alternative, DOE would send all current and future spent nuclear fuel inventories from both DOE and the Naval Nuclear Propulsion Program to one DOE site for interim storage until final disposition.

The two options under Alternative 5 encompass the extreme ranges of spent nuclear fuel inventories that DOE could store at the INEL (i.e., all or none of the inventory). Under option 5a, DOE would ship the INEL spent nuclear fuel inventory off the site to the Hanford Site, the Savannah River Site, the Nevada Test Site, or the Oak Ridge Reservation. Under option 5b, DOE would ship all existing spent nuclear fuel to the INEL.

This alternative would bound the maximum number of spent nuclear fuel-related actions that DOE could reasonably undertake at any site. DOE would have to build new facilities at the selected site to accommodate the increased inventories. Shipments of spent nuclear fuel to the sites not selected as the centralized destination would continue as an interim action pending the construction of necessary storage and examination facilities at the selected site. DOE would then transfer all spent nuclear fuel to the selected site, and the other sites would close their spent nuclear fuel facilities. Before DOE would ship spent nuclear fuel from the originating site, it would characterize and can all spent nuclear fuel as necessary.

The locations from which spent nuclear fuel would originate, in addition to the Hanford Site and Savannah River Site, would include Argonne National Laboratory - East, Babcock and Wilcox, Brookhaven National Laboratory, General Atomics, Los Alamos National Laboratory, Oak Ridge National Laboratory, Sandia National Laboratories, West Valley, and Fort St. Vrain. This alternative would also include fuel that might be returned to the United States following irradiation or testing.

This alternative would include activities related to the treatment of spent nuclear fuel, including research and development and pilot programs to support future decisions on its disposition. DOE would use historic data on spent nuclear fuel to provide a foundation case for determining the impacts associated with potential pilot program activities.

3.1.5.1 Alternative 5a - Centralization at Other DOE Sites.

3.1.5.1.1 Transportation - This option assumes that the INEL would consolidate and prepare all existing and projected onsite spent nuclear fuel for shipment to another DOE facility: the Hanford Site, the Savannah River Site, the Nevada Test Site, or Oak Ridge.

3.1.5.1.2 Stabilization - The DOE would construct a canning and characterization facility at the Idaho Chemical Processing Plant to accept the different types of INEL spent nuclear fuel in various shipping casks and storage containers, and to stabilize these fuel types before their shipment to the selected DOE facility.

3.1.5.1.3 Storage - As in Alternative 1, DOE would complete the CPP-603 Underwater Fuel Storage Facility pool inventory transfer to existing dry storage facilities by 2000. DOE would not build the Dry Fuels Storage Facility. DOE would then close all spent nuclear fuel-related facilities at the INEL with the exception of those in direct support of operating reactors, such as the Advanced Test Reactor canal or the Argonne National Laboratory-West Hot Fuel Examination Facility and Fuel Cycle Facility. This closure would require the establishment of a major surveillance and maintenance operation until DOE determined the disposition of these facilities. The timeframe for closure would depend on the following factors:

- The time necessary to stabilize the spent nuclear fuel in the CPP-603 Underwater Fuel Storage Facility

- The time necessary for the selected DOE site to prepare facilities qualified to accept the spent nuclear fuel

- The time necessary for the procurement and licensing of shipping containers that would be compatible with the selected receiving DOE site

The spent nuclear fuel inventory that DOE would export off the INEL site for Alternative 5a is the same quantity listed for Alternative 1 (see Table 3-3).

3.1.5.1.4 Research and Development - Under this option there would be a phaseout of all research and development activities, although the Electrometallurgical Process Demonstration Project would continue at the Argonne National Laboratory - West Fuel Cycle Facility (but would stabilize only spent nuclear fuel currently on the site).

3.1.5.1.5 Naval-Type Fuel Examination - As with Alternative 1, DOE would phase out shipments of naval-type spent nuclear fuel to the INEL and would phase out the Expended Core Facility.

3.1.5.2 Alternative 5b - Centralization at the INEL.

3.1.5.2.1 Transportation - This option assumes that the INEL would receive all DOE and naval-type spent nuclear fuel (see Table 3-3).

3.1.5.2.2 Stabilization - The Hanford Site, the Savannah River Site, and other DOE facilities would stabilize as necessary, spent nuclear fuel for safe transportation to the Idaho Chemical Processing Plant. The Hanford Site, the Savannah River Site, and other DOE facilities would procure an undetermined number of additional casks and install cask handling equipment as necessary. DOE would complete an expanded Dry Fuels Storage Facility at the INEL, which would include a new Canning and Characterization Facility similar to that described for Alternative 3. This facility would, if needed, repackage the spent nuclear fuel into compatible canisters for dry storage. Other new facility projects would be the same as those described for Alternative 3. In addition, DOE would begin stabilizing for safe storage all complex-wide spent nuclear fuel, as necessary, in existing facilities at the Idaho Chemical Processing Plant. Upgrades and new facilities would be necessary to support long-term fuel stabilization for ultimate disposition; this would address criticality (unplanned and uncontrolled nuclear fission) concerns about the disposal of spent nuclear fuel in a potential Federal repository.

3.1.5.2.3 Storage - Projects and activities for storage of spent nuclear fuel would be similar to those described for Alternative 3, except that accelerated schedules for the Increased Rack Capacity and Additional Increased Rack Capacity projects would be necessary to accommodate the increased fuel receipts. In addition, the schedule for the Dry Fuel Storage Facility project would have to be accelerated and its scope expanded. For example, the Increased Rack Capacity project may have to be completed in late 1996, the Additional Increased Rack Capacity project may have to be completed in late 1998, and the Expanded Dry Fuels Storage Facility project may have to be completed in 2002. If the Expanded Dry Fuels Storage Facility would become available even earlier, it could eliminate the need for the Additional Increased Rack Capacity project.

3.1.5.2.4 Research and Development - DOE would conduct maximum spent nuclear fuel research and development under this option. As with Alternative 4, the Electrometallurgical Process Demonstration Project would continue at the Argonne National Laboratory - West.

3.1.5.2.5 Naval-Type Fuel Examination - Similar to Alternative 3, the practice of transporting spent nuclear fuel from naval reactors to the Expended Core Facility at the INEL would resume.

3.2 Comparison of Alternatives

Chapter 5 analyzes the environmental consequences of the alternatives. Tables 3-4 through 3-6 summarize and compare the potential impacts associated with each alternative from the information in Chapter 5 for construction, normal operations, and accidents, respectively.

A review of the impacts of the alternatives, as presented in Chapter 5, indicates that impacts would be minimal or negligible in most areas. Further, most areas with measurable impacts would have no appreciable differences among alternatives.

In general, the levels of potential impacts associated with Alternatives 1 through 4 (option 4a) would be similar because the amounts of spent nuclear fuel that DOE would manage at the INEL under these alternatives would be on the same order of magnitude (e.g., 300 to 450 MTHM) and activities would extend throughout the full 40-year management period. The lowest level of overall potential impact at the INEL would occur under Alternative 4b(2) - Regionalization by Geography (Elsewhere) and Alternative 5a - Centralization at Other DOE Sites because DOE would ship INEL spent nuclear fuel off the site well before the management period ended in 2035. Alternative 5b and Alternative 4b(1), under which DOE would ship all or nearly all spent nuclear fuel to the INEL, would result in the greatest potential onsite impacts.

Table 3-4. Comparison of impacts from construction.

Area of Impact	1. No Action	2. Decentralization	3. 1992/1993 Planning Basis	4a.[a] Regionalization by Fuel Type	5a. Centralization at Other DOE Sites	5b. Centralization at the INEL
Land Use	No adverse impacts; construction on 0.8 acre[c] in previously disturbed area.	Same as No-Action Alternative	No adverse impacts; construction on 19.3 acres in previously disturbed area.	Same as Alternative 3	Same as No-Action Alternative	No adverse impacts; construction on 30.8 acres in previously disturbed area.
Socioeconomics	No impacts; no net change in employment.	Same as No-Action Alternative	Temporary positive impact on employment with the creation of approximately 375 jobs (peak).	Same as Alternative 3	Temporary positive impact on employment with the creation of approximately 50 jobs (peak).	Same as Alternative 3
Cultural Resources	No adverse impacts; area has been surveyed.	Same as No-Action Alternative	Potential impacts to historic structure; would be mitigated as appropriate.	Same as Alternative 3	Same as No-Action Alternative	Same as Alternative 3
Aesthetic and Scenic Resources	No adverse impacts; previously disturbed areas.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
Geologic Resources	Minor localized impacts; consumption of approximately 158,000 cubic meters[b] of aggregate onsite.	Same as No-Action Alternative	Minor localized impacts; consumption of approximately 392,000 cubic meters of aggregate onsite.	Same as Alternative 3	Same as No-Action Alternative	Minor localized impacts; consumption of approximately 1,772,000 cubic meters of aggregate onsite.

Table 3-4. (continued).

Area of Impact	1. No Action	2. Decentralization	3. 1992/1993 Planning Basis	4a.[a] Regionalization by Fuel Type	5a. Centralization at Other DOE Sites	5b. Centralization at the INEL
Air Quality	Nonradiological: Temporary and intermittent increases in fugitive airborne dust and in exhaust emissions from support equipment. Estimated air quality impacts would be well below established Federal and state standards.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
	Radiological: No radiological impacts from construction activities.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
Water Quality	No adverse offsite impacts to either surface water or groundwater.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
Ecological Resources	Temporary minor impacts; construction confined to previously disturbed areas.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	No impacts	Minimal impacts; construction activities would temporarily disturb wildlife.
Noise	Potential temporary increase in ambient noise levels in construction areas; no change in traffic noise levels.	Same as No-Action Alternative	Potential temporary increase in ambient noise levels in construction areas; small change in traffic noise levels but no change in community reaction to noise.	Same as Alternative 3	Same as Alternative 3	Same as Alternative 3

Table 3-4. (continued).

Area of Impact	1. No Action	2. Decentralization	3. 1992/1993 Planning Basis	4a.[a] Regionalization by Fuel Type	5a. Centralization at Other DOE Sites	5b. Centralization at the INEL
Traffic and Transportation	Negligible impact on traffic.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
Occupational and Public Health and Safety	Occupational: Small occupational radiation exposures within INEL guidance.	Same as No-Action Alternative	Same as No-Action Alternative except 23 potential injuries/ illnesses for construction workers.	Same as Alternative 3	Same as No-Action Alternative except 3 potential injuries/ illnesses for construction workers.	Same as No-Action Alternative except 23 potential injuries/illnesses for construction workers.
	Public: No impact.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
INEL Services	No adverse impacts; modest changes that would be easily accommodated.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
Materials and Waste Management	9 cubic meters[b] of industrial and commercial solid waste from 1995 through 1996.	Same as No-Action Alternative	Cumulative total of 620 cubic meters of industrial and commercial solid waste, 1,500 cubic meters of low-level waste would be generated from 1995 through 1999.	Same as Alternative 3	Cumulative total of 50 cubic meters of industrial and commercial solid waste.	Cumulative total of 3,800 cubic meters of industrial and commercial solid waste and 1,500 cubic meters of low-level waste would be generated from 1995 through 2008.

a. The data provided are for Alternative 4a. Alternative 4b(1) data are the same as those for Alternative 5b. Alternative 4b(2) data are the same as those for Alternative 5a.
b. To convert cubic meters to cubic feet, multiply by 35.3.
c. To convert acres to square kilometers, multiply by 0.004.

Table 3-5. Comparison of impacts from normal operations.

Area of Impact	1. No Action	2. Decentralization	3. 1992/1993 Planning Basis	4a.[a] Regionalization by Fuel Type	5a. Centralization at Other DOE Sites	5b. Centralization at the INEL
Socioeconomics	No impact; no net change in employment.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
Air Quality	Nonradiological: Potential contribution to ambient concentrations would be below applicable standards and regulations.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
	Radiological: Worker doses, doses to the maximally exposed individual, and population dose would be negligible.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
Water Quality	No adverse offsite impacts to either surface water or groundwater.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
Ecological Resources	Negligible impacts, primarily due to continued exclusion of plants and animals from existing facility areas.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Minimal impacts due to generally increased level of operational activity.
Noise	Small change in ambient noise levels in operational areas; no change in traffic noise level.	Same as No-Action Alternative	Small change in ambient noise levels in operational areas; small change in traffic noise levels but no change in community reaction to noise.	Same as Alternative 3	Same as Alternative 3	Same as Alternative 3

Table 3-5. (continued).

Area of Impact	1. No Action	2. Decentralization	3. 1992/1993 Planning Basis	4a.[a] Regionalization by Fuel Type	5a. Centralization at Other DOE Sites	5b. Centralization at the INEL
Traffic and Transportation	Occupational radiation impact: 1.4×10^{-3} LCFs[b] over 40 years.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
	Public radiation impact: 4.4×10^{-5} LCFs over 40 years.	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
Occupational and Public Health and Safety	Occupational radiation impact: 4×10^{-4} LCFs over 40 years.	Occupational radiation impact: 4×10^{-4} LCFs over 40 years.	Occupational radiation impact: 8×10^{-2} LCFs over 40 years.	Same as Alternative 3	Occupational radiation impact: 4×10^{-2} LCFs over 40 years.	Occupational radiation impact: 8×10^{-1} LCFs over 40 years.
	Public radiation impact: 2×10^{-3} LCFs over 40 years.	Public radiation impact: 2×10^{-3} LCFs over 40 years.	Public radiation impact: 4×10^{-3} LCFs over 40 years.	Public radiation impact: 4×10^{-3} LCFs over 40 years.	Public radiation impact: 2×10^{-3} LCFs over 40 years	Public radiation impact: 8×10^{-3} LCFs over 40 years.
INEL Services	Less than 0.1 percent increase in electricity demand and approximately 0.25 percent increase in fuel oil consumption. No increases in water consumption or wastewater generation.	Same as No-Action Alternative	Approximately 1 percent increase in electricity demand and 3 percent increase in fuel oil consumption, which are well within current system capacities or usage limits. No increase in water consumption or wastewater generation.	Same as Alternative 3	Approximately 1.0 percent increase in electricity demand and 2.7 percent increase in fuel oil consumption, which are well within current system capacities or usage limits. No increase in water consumption or wastewater generation.	Approximately 5.3 percent increase in electricity demand, 0.7 percent increase in water consumption, negligible increase in wastewater generation, and 9.7 percent increase in fuel oil consumption, which are well within current system capacities or usage limits.

Table 3-5. (continued).

Area of Impact	1. No Action	2. Decentralization	3. 1992/1993 Planning Basis	4a.[a] Regionalization by Fuel Type	5a. Centralization at Other DOE Sites	5b. Centralization at the INEL
Materials and Waste Management	No increase in waste generation.	Same as No-Action Alternative	Waste generation would increase annually as follows: Industrial and commercial solid waste - 600 cubic meters[c] from 1996 through 2035. Low-level waste - 200 cubic meters from 1996 through 2035. High-level waste - 3 cubic meters from 1996 through 2024. Mixed low-level waste - <1 cubic meters from 1996 through 2024. Transuranic waste - 32 cubic meters from 1996 through 2024.	Same as Alternative 3	Waste generation would increase annually as follows: Industrial and commercial solid waste - 210 cubic meters from 1996 through 2024. Low-level waste - 83 cubic meters from 1996 through 2024. High-level waste, mixed low-level waste, and transuranic waste - same as Alternative 3.	Waste generation would increase annually as follows: Industrial and commercial solid waste - 2,600 cubic meters from 1996 through 2035. Low-level waste - 410 cubic meters from 1996 through 2035. High-level waste - 120 cubic meters from 1996 through 2034. Mixed low-level waste and transuranic waste - same as Alternative 3.

a. The data provided are for Alternative 4a. Alternative 4b(1) data are the same as those for Alternative 5b. Alternative 4b(2) data are the same as those for Alternative 5a.
b. To convert cubic meters to cubic feet, multiply by 35.3.
c. LCFs = Latent Cancer Fatalities.

Table 3-6. Comparison of impacts from accidents.

Area of Impact	1. No Action	2. Decentralization	3. 1992/1993 Planning Basis	4a.[a] Regionalization by Fuel Type	5a. Centralization at Other DOE Sites	5b. Centralization at the INEL
Facility Accidents (Maximum reasonably foreseeable accident[c])	Individual Worker Radiological Risk[b]: 1.8×10^{-10} LCFs[d]/year	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	3.6×10^{-6} LCFs/year
	Public (Population) Radiological Risk[a]: 7.0×10^{-5} LCFs/year	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
Transportation Accident (Maximum reasonably foreseeable accident)	Public (Population) Radiological Risk: 1.1×10^{-5} LCFs/year[d]	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
	Occupational Traffic Fatalities over 40 years: 7.1×10^{-4}	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative
	Public Traffic Fatalities over 40 years: 2.5×10^{-3}	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative	Same as No-Action Alternative

a. The data provided are for Alternative 4a. Alternative 4b(1) data are the same as those for Alternative 5b. Alternative 4b(2) data are the same as those for Alternative 5a.

b. Risk is the product of accident probability and consequences (latent cancers fatalities).

c. This accident represents the maximum reasonably forseeable accident analyzed with the largest consequences to the receptor.

d. LCFs = Latent Cancer Fatalities.

e. Includes noninvolved INEL worker population downwind of the accident; INEL workers are a small portion of the affected population.

4. AFFECTED ENVIRONMENT

4.1 Overview

Chapter 4 describes the existing environment at the Idaho National Engineering Laboratory (INEL) site and the surrounding region. It emphasizes areas that the proposed spent nuclear fuel management alternatives could affect. The information in this chapter provides the existing environmental conditions against which the Department of Energy (DOE) can measure the potential environmental effects of the alternatives. It supports the assessment of the potential environmental consequences that Chapter 5 discusses. DOE used the discussion of the Affected Environment in Volume 2 of this EIS as input for this chapter.

4.2 Land Use

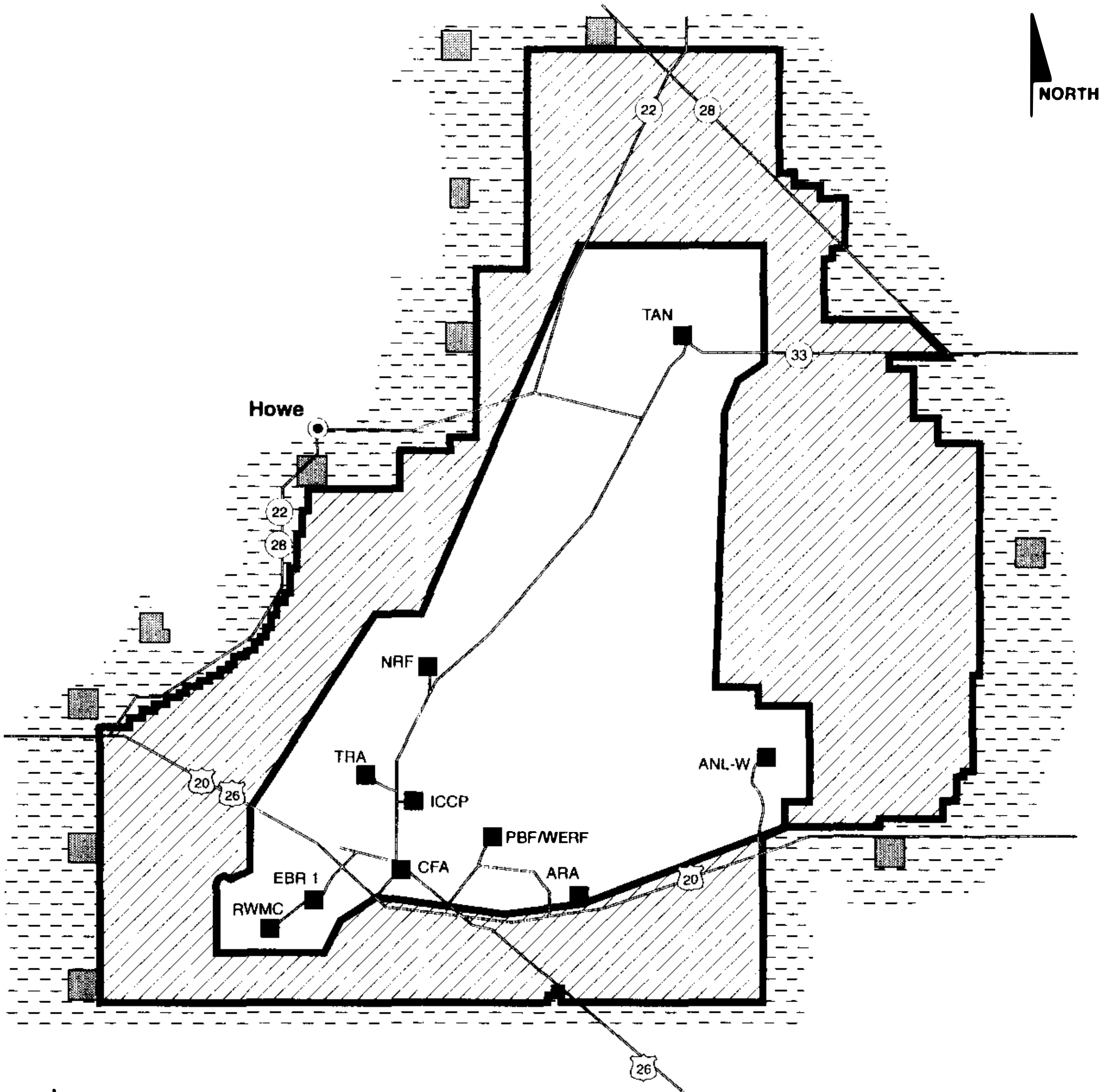
The INEL site encompasses 570,914 acres (2,310.4 square kilometers) in Butte, Bingham, Jefferson, Bonneville, and Clark Counties, Idaho. This section describes existing land uses at the INEL and in the surrounding region, and land use plans and policies applicable to the surrounding area.

4.2.1 Existing and Planned Land Uses at the INEL

Categories of land use at the INEL include facility operations, grazing, general open space, and infrastructure such as roads. Facility operations include industrial and support operations associated with energy research and waste management activities (DOE also conducts such activities at its Idaho Falls facilities). In addition, DOE uses INEL land for recreation and environmental research associated with the designation of the INEL as a National Environmental Research Park.

Much of the INEL is open space that DOE has not designated for specific uses. Some of this open space serves as a buffer zone between INEL facilities and other land uses. Facilities and operations use about 2 percent of the total INEL site area (11,400 acres or 46 square kilometers). Public access to most facility areas is restricted. Approximately 6 percent of the INEL, or 32,985 acres (133.5 square kilometers), is devoted to public roads and utility rights-of-way that cross the site. Recreational uses include public tours of general facility areas and the Experimental Breeder Reactor-I (a National Historic Landmark), and controlled hunting, which is generally restricted to 0.5 mile (0.8 kilometer) inside the INEL boundary.

Cattle and sheep grazing occupies between 300,000 and 350,000 acres (1,200 and 1,400 square kilometers). The U.S. Sheep Experiment Station uses a 900-acre (3.6-square-kilometer) portion of this land, at the junction of Idaho State Highways 28 and 33, for a winter feed lot for approximately 6,500 sheep. Grazing is not allowed within 2 miles (3.2 kilometers) of any nuclear facility and, to avoid the possibility of milk contamination by long-lived radionuclides, dairy cattle are not pennitted on the site. The Department of the Interior's Bureau of Land Management grants and administers rights-of-way and grazing permits. Figure 4.2-1 shows selected land uses at the INEL and in the surrounding region.



Legend:

ANL-W	Argonne National Laboratory-West
ARA	Auxiliary Reactor Area
CFA	Central Facilities Area
EBR-I	Experimental Breeder Reactor - I
ICPP	Idaho Chemical Processing Plant
NRF	Naval Reactors Facility
PBF	Power Burst Facility
RWMC	Radioactive Waste Management Complex
TAN	Test Area North
TRA	Test Reactor Area
WERF	Waste Experimental Reduction Facility

Under grazing permits

BLM or private land

State land

Miles 0 2 4 6 8

Kilometers 0 4 8 12

PJ20-3

Figure 4.2-1. Selected land uses at the INEL and in the surrounding region.

The INEL site is within the Medicine Lodge Resource Area (approximately 140,415 acres or 568.3 square kilometers in the eastern and southern portions of the INEL site) and the Big Butte Resource Area (430,499 acres or 1,742 square kilometers in the central and western portions); the Bureau of Land Management administers both of these areas. Under Resource Management Plans, the Bureau manages portions of these Resource Areas for grazing and wildlife habitat. No mineral exploration or development is allowed on INEL land.

DOE land use plans and policies applicable to the INEL include the INEL *Institutional Plan - Fiscal Year 1994 - 1999* (DOE-ID 1993c) and the *INEL Technical Site Information Report* (DOE-ID 1993a). The *Institutional Plan* provides a general overview of INEL facilities, outlines strategic program directions and major construction projects, and identifies specific technical programs and capital equipment needs. The *Technical Site Information Report* presents a 20-year master plan for development activities at the site. Under the scope of these planning documents, energy research and waste management activities would continue in existing facility areas and, in some instances, expand into currently undeveloped site areas. These documents also describe environmental restoration, waste management, and spent nuclear fuel activities. Projected land use scenarios for the next 25 to 50 years include the outgrowth of current functional areas and the possible development of waterfowl production ponds in existing grazing areas.

No onsite land use restrictions due to Native American treaty rights would exist for any of the alternatives described in this EIS. The INEL does not lie within any of the land boundaries established by the Fort Bridger Treaty, and the entire INEL site is land occupied by the U.S. Department of Energy. Therefore, the provisions in the Fort Bridger Treaty that allows the Shoshone-Bannock Indians to hunt on unoccupied lands of the United States do not apply to the INEL site.

4.2.2 Existing and Planned Land Use in Surrounding Areas

The Federal government, the State of Idaho, and private parties own the lands surrounding the INEL site. Land uses on Federally owned land consist of grazing, wildlife management, range land, mineral and energy production, and recreational uses. State-owned lands are used for grazing, wildlife management, and recreational purposes. Privately owned lands are used primarily for grazing, crop production, and range land.

Small communities and towns near the INEL boundaries include Mud Lake to the east; Arco, Butte City, and Howe to the west; and Atomic City to the south. The larger communities of Idaho Falls, Rexburg, Blackfoot, and Pocatello and Chubbock are to the east and southeast of the INEL site. The Fort Hall Indian Reservation is to the southeast of the INEL. Recreation and tourist attractions in the region around the INEL include the Craters of the Moon National Monument, Hell's Half Acre Wilderness Study Area, Black Canyon Wilderness Study Area, Camas National Wildlife Refuge, Market Lake State Wildlife Management Area, North Lake State Wildlife Management Area, Yellowstone National Park, Grand Teton National Park, Jackson Hole Recreation Complex, Targhee and Challis National Forests, and the Snake River.

Lands surrounding the INEL site are subject to Federal and state planning laws and regulations. Federal rules and regulations that require public involvement in their implementation govern planning for and use of Federal lands and their resources. Land use planning in the State of Idaho is derived from the Local Planning Act of 1975 (State of Idaho Code 1975). Because the State currently has no land use planning agency, the Idaho legislature requires each county to adopt its own land use planning and zoning guidelines. County plans that are applicable to lands bordering the INEL site include the Clark County Planning and Zoning Ordinance and Interim Land Use Plan (Clark County 1994); Bonneville County Comprehensive Plan (Bonneville County 1976); Bingham County Zoning Ordinance and Planning Handbook (Bingham County 1986); Jefferson County Comprehensive Plan (Jefferson County 1988); and Butte County Comprehensive Plan (Butte County 1992). Land use planning for INEL facilities within the Idaho Falls city limits is subject to Idaho Falls planning and zoning restrictions (City of Idaho Falls 1989, 1992).

All county plans and policies accept development adjacent to previously developed areas to minimize the need to extend infrastructure improvements and to avoid urban sprawl. Because the INEL is remote from most developed areas, INEL lands and adjacent areas are not likely to experience residential and commercial development; no new development is planned near the INEL site. However, DOE expects recreational and agricultural uses to increase in the surrounding area in response to greater demand for recreational areas and the conversion of range land to crop land.

4.3 Socioeconomics

This section presents a brief overview of current socioeconomic conditions within a region of influence where approximately 97 percent of the INEL workforce lived in 1991 (DOE-ID 1991). The INEL region of influence is a seven-county area comprised of Bingham, Bonneville, Butte, Clark, Jefferson, Bannock, and Madison Counties. The region of influence also includes the Fort Hall Indian Reservation and Trust Lands (home of the Shoshone-Bannock Tribes) in Bannock, Bingham, Caribou, and Power Counties.

4.3.1 Employment

Historically, the regional economy has relied predominantly on natural resource use and extraction. Today, farming, ranching, and mining remain important components of the regional economy. Idaho Falls is the retail and service center for the region of influence, and Pocatello has evolved into an important processing and distribution center and site of higher education institutions.

4.3.1.1 Region. The labor force in the region of influence increased from 92,159 in 1980 to 104,654 in 1991, an average annual growth rate of approximately 1.2 percent. In 1991 the region of influence accounted for approximately 18 percent of the total state labor force of 504,000 (ISDE 1992). As listed in Table 4.3-1, the projected labor force in the region of influence will reach 108,667 by 1995.

Unemployment rates varied considerably among the counties of the region of influence in 1991, ranging from 2.6 percent in Clark County to 6.3 percent in Bannock and Bingham Counties. Since 1980 the average annual unemployment rate for the region has ranged from 5.3 percent in 1989 to 8.3 percent in 1983. In 1991 the average annual unemployment rate for the region of influence was 5.5 percent compared to the statewide average of 6.2 percent (ISDE 1992).

Employment in the region of influence increased from 86,261 in 1980 to 98,898 in 1991, an average annual growth rate of approximately 1.3 percent. As listed in Table 4.3-1, employment is projected to increase to 101,450 by 1995.

Table 4.3-1. Projected labor force, employment, and population for the INEL region of influence, 1995-2004.

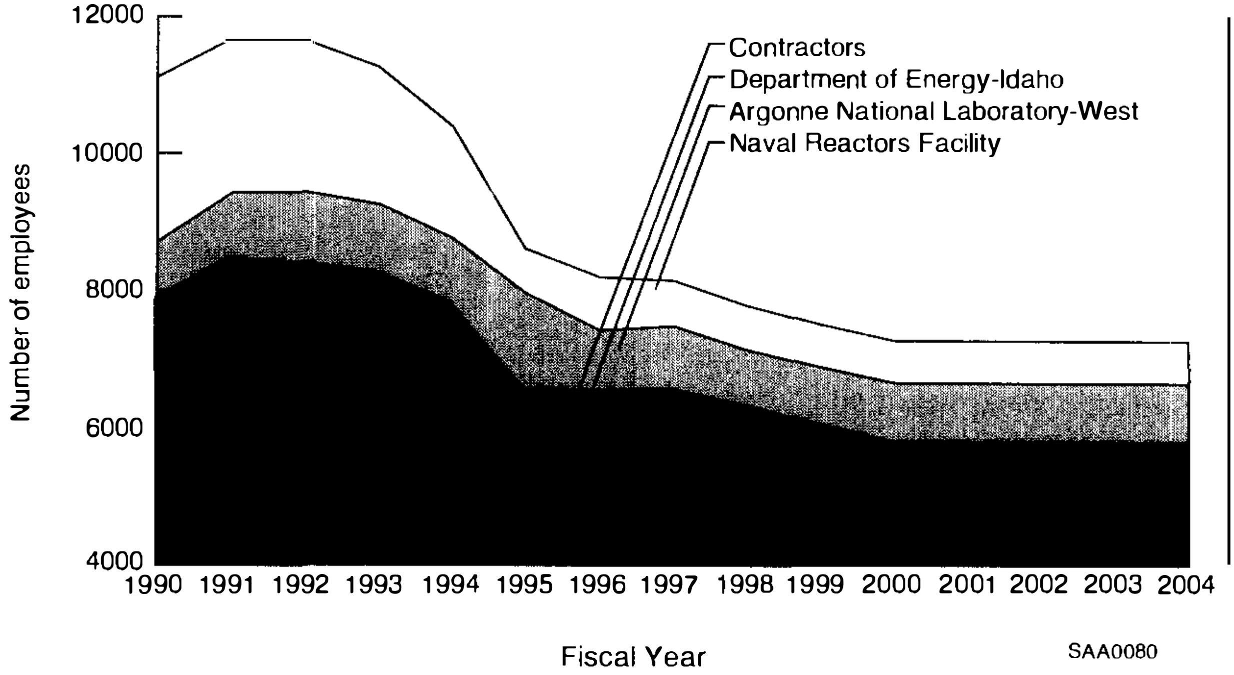
	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Labor Force	108,667	109,607	110,547	111,487	112,427	113,367	114,308	115,248	116,188	117,128
Employment	101,450	102,328	103,205	104,083	104,960	105,838	106,716	107,593	108,471	109,348
Population	247,990	251,518	255,096	258,726	262,406	266,140	268,667	271,219	273,795	276,395

Source: ISDE (1992); SAIC (1994); ISDE (1991); ISDE (1986).

4.3.1.2 Idaho National Engineering Laboratory. INEL plays a substantial role in the regional economy. During Fiscal Year 1990, INEL directly employed approximately 11,100 personnel, accounting for almost 12 percent of total regional employment. The estimated population directly supported by INEL employment was approximately 38,000 persons, or 17 percent of the total regional population. The major employers at INEL are DOE-ID, DOE-ID contractors, Argonne National Laboratory-West, and the Naval Reactors Facility (see Figure 4.3-1). In 1992, the total direct INEL employment was approximately 11,600 jobs (DOE-ID 1994). Projections as of January 1995 indicate that the total number of jobs at INEL will decrease to approximately 8,620 in Fiscal Year 1995 and to approximately 7,250 in Fiscal Year 2004 (Tellez 1995). Projected decreases in INEL employment are primarily related to contractor consolidation, which accounts for 64 percent of the projected losses between Fiscal Year 1994 and Fiscal Year 2004, and to reduced activities at the Naval Reactors Facility, which accounts for 33 percent of the projected job losses. Contract changes at DOE-ID resulted in the consolidation of several contracts under one contract. The consolidation eliminated redundant administrative activities previously performed by each individual contractor and offered early retirement or other options to impacted INEL contractor employees.

4.3.2 Population and Housing

4.3.2.1 Population. From 1960 to 1990, population growth in the region of influence mirrored statewide growth. During this period, the region's population increased at an average annual rate of approximately 1.3 percent, while the growth rate for the State was 1.4 percent. Between 1980 and 1990, population growth in the region of influence approximately equaled that of the State with an average growth rate of 0.6 percent per year. The region of influence had a 1990 population of 219,713, which comprised 22 percent of the total State population of 1,006,749. Based on population and employment trends, the population in the region of influence will reach approximately 248,000 persons by 1995 (Table 4.3-1).



Source: Tellez (1995); DOE-ID (1994)

PJ20-3

Figure 4.3-1. Historic and projected baseline employment at the Idaho National Engineering Laboratory, 1990-2004.

In 1990, the most populous counties were Bannock and Bonneville, which together contained over 60 percent of the seven-county total (Figure 4.3-2). Butte and Clark were the least populous of the counties in the region of influence. The largest cities in the region of influence are Pocatello and Idaho Falls, with 1990 populations of approximately 46,000 and 44,000, respectively. In 1990, the Fort Hall Indian Reservation and Trust Lands contained 5,113 residents, most of whom (52 percent) resided in Bingham County.

4.3.2.2 Housing. Bonneville and Bannock Counties (which respectively include the cities of Idaho Falls and Pocatello) provided 67 percent of the 73,230 year-round housing units in the region of influence in 1990 (see Table 4.3-2). Of this number, approximately 70 percent were single-family units, 17 percent were multifamily units, and 13 percent were mobile homes. Most of the multifamily units (75 percent) were in Bonneville and Bannock Counties. About 29 percent of the occupied housing units in the region were rental units and 71 percent were homeowner units (USBC 1992).

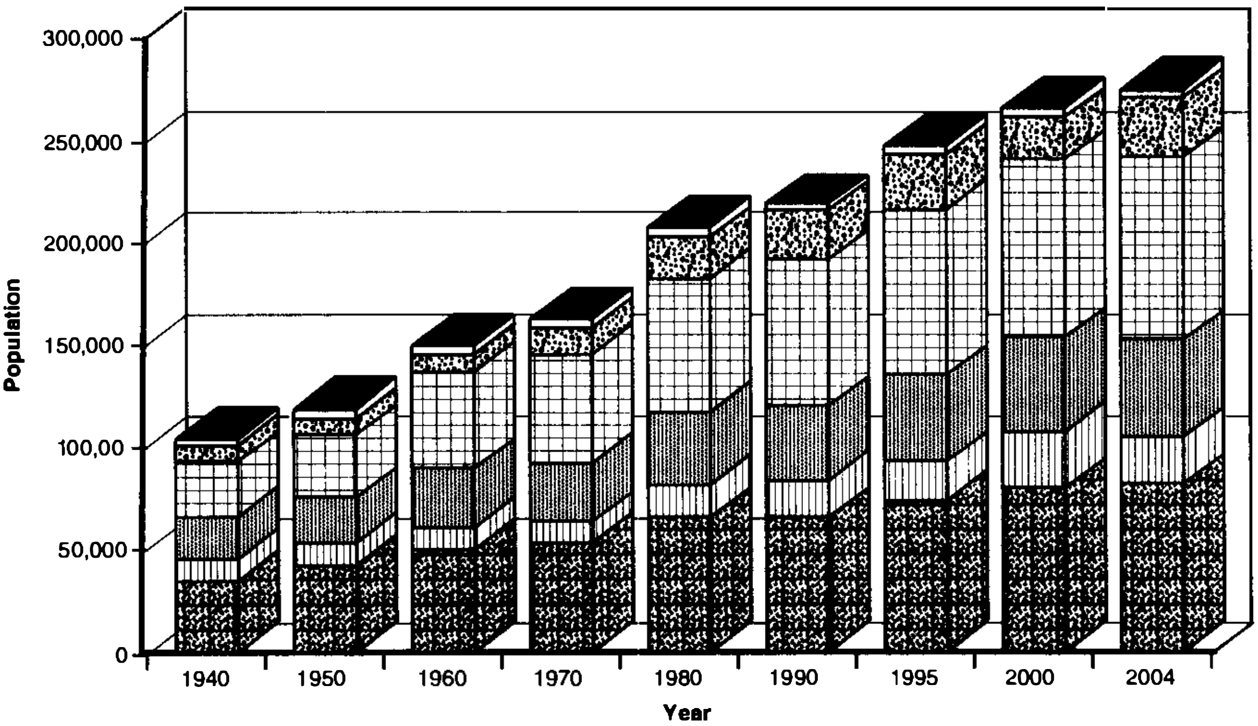
The median value of owner-occupied housing units ranged from $37,300 in Clark County to $68,700 in Madison County, and median monthly rents ranged from $243 in Butte County to $366 in Bonneville County. In 1990, there were 1,510 occupied housing units on the Fort Hall Indian Reservation and Trust Lands (USBC 1992) and a vacancy rate of 14 percent.

4.3.3 Community Services

This assessment considers the following selected community services in the region of influence: public schools, law enforcement, fire protection, hospital services, and solid waste disposal. Table 4.3-3 summarizes pertinent characteristics of these services for the region of influence.

Seventeen public school districts and three nonpublic schools provide educational services for about 58,000 children in the region of influence. Of these students, about 6,500 were dependents of INEL-related employees. During the 1990-1991 academic year, most public school districts spent an average of $3,000 to $4,000 per student annually. Higher education in the region is provided by the University of Idaho, Idaho State University, Brigham Young University, Ricks College, and the Eastern Idaho Technical College.

Seven county sheriff's offices, 12 city police departments, and the Idaho State Police provide law enforcement services in the region. There was a total of 479 sworn officers and 100 other law



Note: 1995 to 2004 represent population projection

Legend:

Clark
Butte
Madison
Bonneville
Bingham
Jefferson
Bannock

Source: USBC (1982); USBC (1992). PJ20-3

Figure 4.3-2. Historic and projected total population for the counties of the region of influence, 1940 through 2004.

Table 4.3-2. Number of housing units, vacancy rates, median house value, and median monthly rent by county and region of influence.[a]

	Homeowner housing units			Rental units		
County	Number of units	Vacancy rates	Median value ($)	Number of units	Vacancy rates	Median monthly rent ($)
Bannock	16,447	2.4	53,300	7,467	10.3	294
Bingham	9,010	2.0	50,700	2,955	9.2	284
Bonneville	17,707	1.9	63,700	7,375	6.2	366
Butte	780	4.6	41,400	302	16.2	243
Clark	177	1.7	37,300	114	9.6	281
Jefferson	4,000	2.0	54,300	992	4.1	314
Madison	3,522	1.3	68,700	2,392	2.8	299
Region of influence	51,674	2.1	-	21,556	4.6	-

a. Source: USBC (1992).

enforcement personnel in 1991, more than 59 percent of whom served Bannock and Bonneville Counties.

Eighteen fire districts in the region of influence operate 30 fire stations staffed by 180 paid and approximately 300 volunteer firefighters. Bingham, Bonneville, Butte, Clark, and Jefferson Counties, which surround the INEL, have developed emergency plans to be implemented in the event of a radiological or hazardous materials emergency. Each emergency plan identifies facilities with extremely hazardous substances and defines transportation routes for these substances. The emergency plans also include procedures for notification and response, listings of emergency equipment and facilities, evacuation routes, and training programs.

Eight hospitals serve the region of influence with more than 900 licensed beds and a capacity of nearly 128,000 patient-days per year. Occupancy rates range from 22.0 to 61.7 percent in the region (IDHW 1990). County governments and the Blackfoot, Dubois, Idaho Falls, and Pocatello fire departments provide regional ambulance services. A private ambulance company serves residents in Butte County. Four quick-response units, two medical helicopters, and two clinics specializing in emergency medical services also serve the region of influence (Hardinger 1990; U.S. West Directories 1992).

Table 4.3-3. Summary of public services available in the region of influence.[a]

Public Service	Bannock	Bingham	Bonneville	Butte	Clark	Jefferson	Madison
	County						
Schools							
Number of public school districts	2	5	3	1	1	3	2
Total enrollment	15,455	11,311	17,896	765	166	5,339	5,967
Number of INEL-related students (excluding military)	485	1,532	4,040	301	5	134	47
Health Care Delivery							
Number of hospitals	3	2	1	1	0	0	1
Number of licensed beds	309	238	311	4	-	-	52
Law Enforcement							
Number of sworn law enforcement officers	151	65	143	4	2	18	43
Total personnel per 1000 population	2.5	2.0	2.2	1.3	6.3	1.6	1.9
Fire Protection							
Number of fire stations	9	7	6	2	1	4	1
Number of firefighters	166	96	121	15	7	63	24
Number of firefighting vehicles	37	25	24	3	1	11	6
Municipal Solid Waste Disposal							
Number of landfills meeting EPA[b] regulations	1[c]	3[d]	1[e]	2	0[f]	1	0[f]
Expected lifespan in years	30	3-6	50	30	-	2	-

a. Source: IDE (1991); IDHW (1990); IDLE (1991); Kouris (1992a); and Kouris (1992b).
b. EPA = U.S. Environmental Protection Agency.
c. Fort Hall Mine Landfill is being redesigned to meet EPA standards.
d. Aberdeen Landfill may close due to noncompliance with EPA standards.
e. A new landfill is replacing Bonneville County Landfill.
f. Madison and Clark Counties are evaluating a regional landfill for use after 1993.

Municipal solid waste generated in the region of influence is transported to county landfills. In 1992, twelve landfills served the region of influence. Four landfills (one each in Bannock, Clark, Jefferson, and Madison Counties) will close without replacement before reaching their planned capacity due to noncompliance with new Environmental Protection Agency standards (CFR 1991a).

4.3.4 Public Finance

In Fiscal Year 1991, total county revenues for the region of influence amounted to approximately $90 million (see Table 4.3-4). County governments receive most of their revenues from taxes and intergovernmental transfers. In 1991 the total assessed value of taxable property in the region of influence was about $4.5 billion. In addition to property tax revenues, local governments (cities and counties) also receive revenue from sales tax disbursements and revenue-sharing programs. These two sources provide approximately 60 to 85 percent of the total revenues received by each county.

Table 4.3-4. Total revenues and expenditures by county, Fiscal Year 1991.[a]

County	Total revenues ($)	Total expenditures ($)
Bannock	16,232,274	14,216,708
Bingham	11,434,200	10,708,011
Bonneville[b]	50,186,650	51,850,100
Butte	1,417,684	1,397,012
Clark	1,236,849	1,086,379
Jefferson	4,408,236	4,566,074
Madison	5,249,432	5,662,080
Seven-county region	90,165,325	89,486,364

a. Sources: Ghan (1992); Bingham County (circa 1992); McFadden (circa 1992); Swager & Swager (1992a); Swager & Swager (1992b); Draney, Searle, and Associates (1992); Schwendiman & Sutton (1992).
b. Bonneville County's financial statements and total revenue data include special accounts for schools, cities, cemeteries, fire districts, ambulance districts, and other special accounts not found in other county budgets. The majority of intergovernmental revenue is used to fund these accounts.

Although DOE as a Federal agency is exempt from paying state or local taxes, INEL employees and contractors are not. In 1992, INEL employees paid an estimated $60 million in Federal withholding tax and $24 million in state withholding tax.

In 1991 the major categories of county government expenditures were general government services, 27 percent; road maintenance, 18 percent; public safety, 16 percent; health and welfare programs, 16 percent; sanitation and public works, 9 percent; debt service, 3 percent; trust remittances, 2 percent; and other expenditures, 9 percent.

4.4 Cultural Resources

This section discusses cultural resources at the INEL, including prehistoric and historic archeological sites and historic sites and structures, and traditional resources that are of cultural or religious importance to local Native Americans. It also discusses paleontological localities on the INEL site.

4.4.1 Archeological Sites and Historic Structures

As summarized in the INEL Draft Management Plan for Cultural Resources (Miller 1992), the INEL contains a rich and varied inventory of cultural resources. This includes fossil localities that provide an important paleontological context for the region and the many prehistoric archeological sites that are preserved within it. These latter sites, including campsites, lithic workshops, cairns, and hunting blinds, among others, are also an important part of the INEL inventory because they provide information about the activities of aboriginal hunting and gathering groups who inhabited the area for approximately 12,000 years. In addition, archeological sites, pictographs, caves, and many other features of the INEL landscape are also important to contemporary Native American groups for historic, religious, and traditional reasons. Historic sites, including the abandoned town of Powell/Pioneer, a northern spur of the Oregon Trail known as Goodale's Cutoff, many small homesteads, irrigation canals, sheep and cattle camps, and stage and wagon trails, document the use of the area during the late 1800s and early 1900s. Finally, the many scientific and technical facilities inside the INEL boundaries have preserved important information on the historic development of nuclear science in America.

To date, more than 100 cultural resource surveys have been conducted over approximately 4 percent of the area on the INEL site. These surveys, most of which have occurred near major facility areas, have identified 1,506 archeological resources, including 688 prehistoric sites, 38 historic sites, 753 prehistoric isolates, and 27 historic isolates (Miller 1992; Gilbert and Ringe 1993). These numbers do not include architectural properties associated with the creation and operation of the INEL. Until formal significance evaluations (archeological testing and historic records searches) have been completed, all cultural sites in this inventory are considered to be potentially eligible for nomination to the National Register of Historic Places. However, all the isolates have been categorized as unlikely to meet eligibility requirements (Yohe 1993).

Due to the relatively high density of prehistoric sites on the INEL and the need to consider these resources during Federal undertakings, DOE has sponsored a preliminary study, which resulted in the development of a predictive model, to identify areas where densities of sites are highest and where the potential impacts to significant archeological resources, as well as costs of compliance, would increase correspondingly (Ringe 1993). This information provides guidance for INEL project managers in the selection of appropriate areas for new construction. However, it does not take the place of inventories that are required by the National Historic Preservation Act before ground-disturbing projects can start (NHPA 1966 as amended).

The predictive model, constructed using a multivariate statistical technique on environmental variables associated with areas with and without sites, indicates that prehistoric cultural resources appear to be concentrated in association with certain definable physical features of the land. In this context, very high densities of resources are likely to occur along the Big Lost River and Birch Creek, atop buttes, and within craters and caves. The Lemhi Mountains, the Lake Terreton basin, and a 1.75-mile- (2,800-meter-) wide zone along the edge of local lava fields probably contain a fairly high density of sites. Within the extensive flows of basaltic lava and along the low foothills of the Lemhi Mountains, site density is classified as moderate, and the lowest density of prehistoric resources probably occurs in the floodplain of the Big Lost River and the alluvial fans emerging from the Birch Creek Valley, in the sinks, and in the recent Cerro Grande lava flow. However, a classification of low or medium density does not eliminate the possibility that significant resources exist in those areas. Although the predictive model has not been tested, it is useful as a planning guide for defining areas most likely to contain archeological resources based on past surveys.

Although there has been no systematic inventory of historically significant facilities associated with the creation and operation of the INEL, a preliminary study indicated that all INEL facilities will require evaluation (Braun et al. 1993). The Experimental Breeder Reactor-I is a National Historic Landmark listed in the National Register of Historic Places. To date, however, few of the other properties have been formally evaluated for eligibility to the National Register. Memoranda of Agreement between DOE, the Idaho State Historic Preservation Office, and the National Advisory Council on Historic Preservation establish that certain structures at Test Area North (DOE 1993b) and Auxiliary Reactor Area (DOE 1993a) are eligible for nomination, and outline specific techniques for preserving the historic value of the areas in conformance with the requirements of the Historic American Building Survey and the Historic American Engineering Record. Other facilities on the INEL site are likely to require similar efforts if DOE schedules them for major modification, demolition, or abandonment.

4.4.2 Native American Cultural Resources

Because Native American people believe the land is sacred, the entire INEL reserve is culturally important to them. Cultural resources, to the Shoshone-Bannock peoples, include all forms of traditional lifeways and usage of all natural resources. This includes not only prehistoric archeological sites, which are important in a religious or cultural heritage context, but also features of the natural landscape, air, plant, water, or animal resources that might have special significance. These resources may be affected by changes in the visual environment (construction, ground disturbance, or introduction of a foreign element into the setting), dust particles, or by contamination. Geographically, the INEL is included within a large territory once inhabited by and still of importance to the Shoshone-Bannock Tribes. Plant resources used by the Shoshone-Bannock Tribes that are located on or near the INEL site are listed in Table 4.4-1. Areas significant to the tribes would include the buttes, wetlands, sinks, grasslands, juniper woodlands, Birch Creek, and the Big Lost River.

Five Federal laws prompt consultation between Federal agencies and Indian Tribes: the National Environmental Policy Act (NEPA 1969), the National Historic Preservation Act (NHPA 1966 as amended), the American Indian Religious Freedom Act (AIRFA 1978), the Archeological Resources Protection Act (ARPA 1979), and the Native American Graves Protection and Repatriation Act (NAGPRA 1990). In accordance with these directives and in consideration of its Native American Policy (DOE 1990a and DOE 1992a), DOE is developing procedures at the INEL for consultation and coordination with the Shoshone-Bannock Tribes of the Fort Hall Reservation. DOE has committed to additional interaction and exchange of information with the Shoshone-Bannock Tribes, and has outlined this relationship in a formal Working Agreement with these tribes (DOE 1992c). In addition, the Cultural Resources Management Plan for the INEL (Miller 1992) and the curation agreement for permanent storage of archaeological materials will be completed by June 1996. The Cultural Resources Management Plan will define procedures for involving the tribes during the planning stages of project development and the curation agreement will provide for the repatriation of burial goods in accordance with NAGPRA.

4.4.3 Paleontological Resources

There are 31 known fossil localities at the INEL site. Available information suggests that the region has relatively abundant and varied paleontological resources. Preliminary analyses suggest that

Table 4.4-1. Plants used by the Shoshone-Bannock tribes that are located on or near the INEL.

Plant Family	Type of Use	Location	Abundance
Desert Parsley	medicine, food	scattered over site	common
Milkweed	food, tools	roadsides	scattered, uncommon
Sagebrush	medicine, tools	throughout the site	common, abundant
Balsamroot	food, medicine	around buttes	common but scattered
Thistle	food	scattered throughout site	common but scattered
Gumweed	medicine	disturbed areas	common
Sunflower	medicine, food	roadside	common
Dandelion	food, medicine	throughout site	common
Beggar's Ticks	food	disturbed areas throughout site	common, abundant
Tansymustard	food, medicine	disturbed areas	common
Cactus	food	throughout the site	common, abundant
Honeysuckle	food, tools	Big Southern Butte	common on butte
Goosefoot	food	throughout site	common, abundant
Russian Thistle	food	disturbed areas throughout site	common, abundant
Dogwood	food, medicine, tools	Webb Springs, Birch Creek	common where found
Juniper	medicine, food, tools	throughout site	common to abundant
Gooseberry	food	scattered throughout site	common
Mentha arvensis	medicine	Big Lost River	uncommon
Wild onion	food, medicine, dye	throughout site	common
Caloehortus spp.	food	buttes	common
Fireweed	food	throughout site	common
Pine	food, tools, medicine	Big Southern Butte	common on butte
Douglas Fir	medicine	Big Southern Butte	common on butte
Plantain	medicine, food	throughout site	uncommon
Wildrye	food, tools	throughout site	common, abundant
Indian Ricegrass	food	throughout site	common, abundant
Bluegrass	food, medicine	throughout site	common, abundant
Serviceberry	food, tools, medicine	buttes	common where found
Chokeberry	food, medicine, tools, fuel	buttes	common where found
Wood's Rose	food, smoking, medicine, ritual	Big Lost River, Big Southern Butte	common, abundant
Red Raspberry	food, medicine	Big Southern Butte	uncommon
Willow	medicine	throughout site in moist areas	common
Coyote Tobacco	smoking, medicine	Big Lost River, Webb Springs	uncommon
Cattail	food, tools	sinks, outflow from facilities	uncommon

Source: Andersen et al. (1995).

these materials are most likely to occur in association with archeological sites; in areas of basalt flows; in deposits of the Big Lost River, Little Lost River, and Birch Creek; in deposits of Lake Terreton and playas; in some wind and sand deposits; and in sedimentary interbeds or lava tubes within local lava flows (Miller 1992).

4.5 Aesthetic and Scenic Resources

4.5.1 Visual Character of the INEL Site

The Bitterroot, Lemhi, and Lost River mountain ranges border the INEL site on the north and west. Persons can see volcanic buttes near the southern boundary of the INEL from most locations on the site and from the Fort Hall Reservation. Most of the INEL site consists of open undeveloped land, covered predominantly by large sagebrush and grasslands (see Section 4.9). Pasture and irrigated farmland border much of the INEL site (see Section 4.2).

Although the INEL has a master plan, it has not established specific visual resource standards. The nine facility areas on the INEL site are generally of low density, look like commercial or industrial complexes, and are spread across the site. Structures in the facility areas range in height from 10 feet to approximately 100 feet (3 to 30 meters). About 90 miles (145 kilometers) of paved public highway run through the INEL site (see Section 4.11). Although many INEL facilities are visible from these highways, most facilities are located more than 0.5 mile (0.8 kilometer) from public roads.

4.5.2 Scenic Areas

The Craters of the Moon National Monument is about 15 miles (24 kilometers) southwest of the INEL site's western boundary. The Monument is located in a designated Wilderness Area, which must maintain Class I (very high) air quality standards or minimal degradation, as defined by the Clean Air Act (CAA 1990; CFR 1990; CFR 1991b). Under Section 169a of the Clean Air Act, air quality includes visibility and scenic view considerations.

Lands adjacent to the INEL under Bureau of Land Management jurisdiction are Visual Resource Management Class II areas (BLM 1984; BLM 1986), which urge preservation and retention of the existing character of the landscape. Lands inside the INEL boundaries are Class III and IV areas, the most lenient classes in terms of modification. The Bureau of Land Management is considering the Black Canyon Wilderness Study Area, which is adjacent to the INEL, for a Wilderness Area designation (BLM 1986); if approved, this would result in an upgrade from Visual Resource Management Class II to a Class I.

Features of the natural landscape have special significance to the Shoshone-Bannock tribes. The visual environment of the INEL site is within the visual range of Fort Hall Reservation.

4.6 Geology

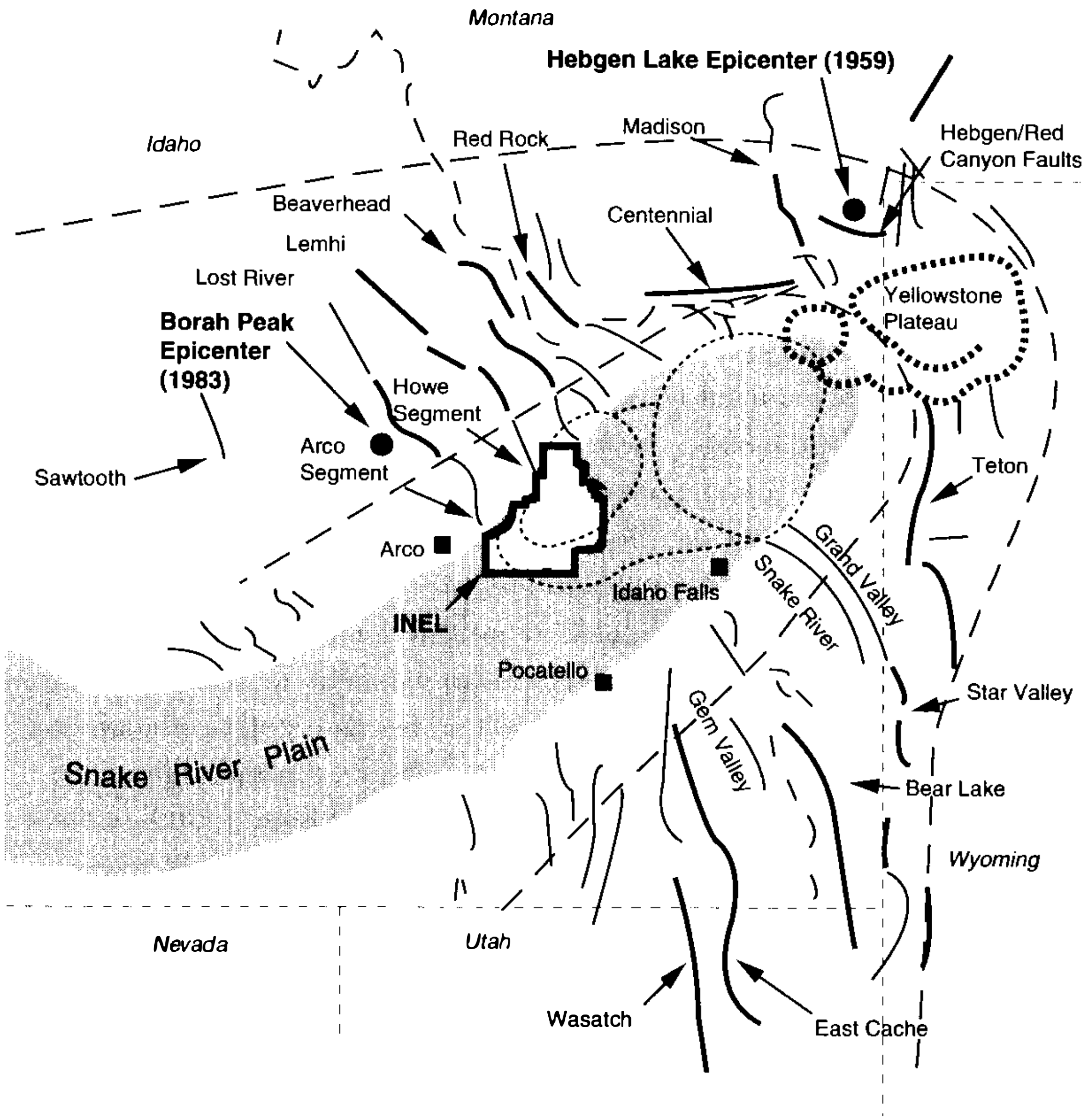
This section describes the geology of the INEL and the surrounding area. Section 4.6.1 characterizes the general geology, while section 4.6.2 describes the natural resources of the area. Sections 4.6.3 and 4.6.4 describe seismic and volcanic hazards, respectively.

4.6.1 General Geology

The site is on the Eastern Snake River Plain (Figure 4.6-1). The Plain forms a broad northeast-trending, crescent-shaped trough with low relief composed primarily of surface basaltic lava flows formed 1.2 million to 2,100 years ago. The Plain features thin, discontinuous, and interbedded deposits of wind-blown loess and sand; water-borne alluvial fan, lacustrine, and floodplain alluvial sediments; and rhyolitic domes formed 1,200,000 to 300,000 years ago (Kuntz et al. 1990) (Figure 4.6-2). Mountains and valleys of the Basin and Range Province, which trend north to northwest and consist of folded and faulted rocks that are more than 70 million years old, bound the Plain on the north and south. The Yellowstone Plateau bounds the Plain on the northeast. The major episode of Basin and Range faulting began 20 to 30 million years ago and continues today, most recently associated with the October 28, 1983, Borah Peak earthquake [moment magnitude 6.9, magnitude 7.3 on the Richter scale with a resulting peak ground acceleration of 0.022 to 0.078 at the INEL (Jackson 1985)], which occurred along the Lost River fault, approximately 100 kilometers (62 miles) from site facilities and the 1959 Hebgen Lake Earthquake, moment magnitude 7.5, approximately 150 kilometers (93 miles) from the INEL (Figure 4.6-1).

The northeast-trending volcanic terrain of the Plain has a markedly different geologic history and tectonic pattern than the folded and faulted terrain of the northwest-trending Basin and Range. The Basin and Range faults have not been observed on or across the Plain. Four northwest-trending volcanic rift zones, attributed to basaltic eruptions that occurred 4 million to 2,100 years ago, lie across the Plain at the INEL (Bowman 1995; Hackett and Smith 1992; Kuntz et al. 1990).

The seismic characteristics of the Eastern Snake River Plain and the adjacent Basin and Range Province are also different. Earthquakes and active faulting are associated with the Basin and Range tectonic activity. The Plain has historically experienced few and small earthquakes (King et al. 1987; Pelton et al. 1990; WCC 1992; Jackson et al. 1993).



Legend:

- ● Large Earthquake Epicenter
- ■ Town
- – — Limits of Parabolic Zone of Seismicity
- ——— Quaternay Normal Faults
- ▬▬▬ Holocene Movement
- ▮▮▮▮▮▮▮ Pleistocene Calderas
- ------- Tertiary Calderas

Source: Map modified from Anders et al. (1989) and Hackett and Morgan (1988)

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Figure 4.6-1. Location of INEL in context of regional geologic features.

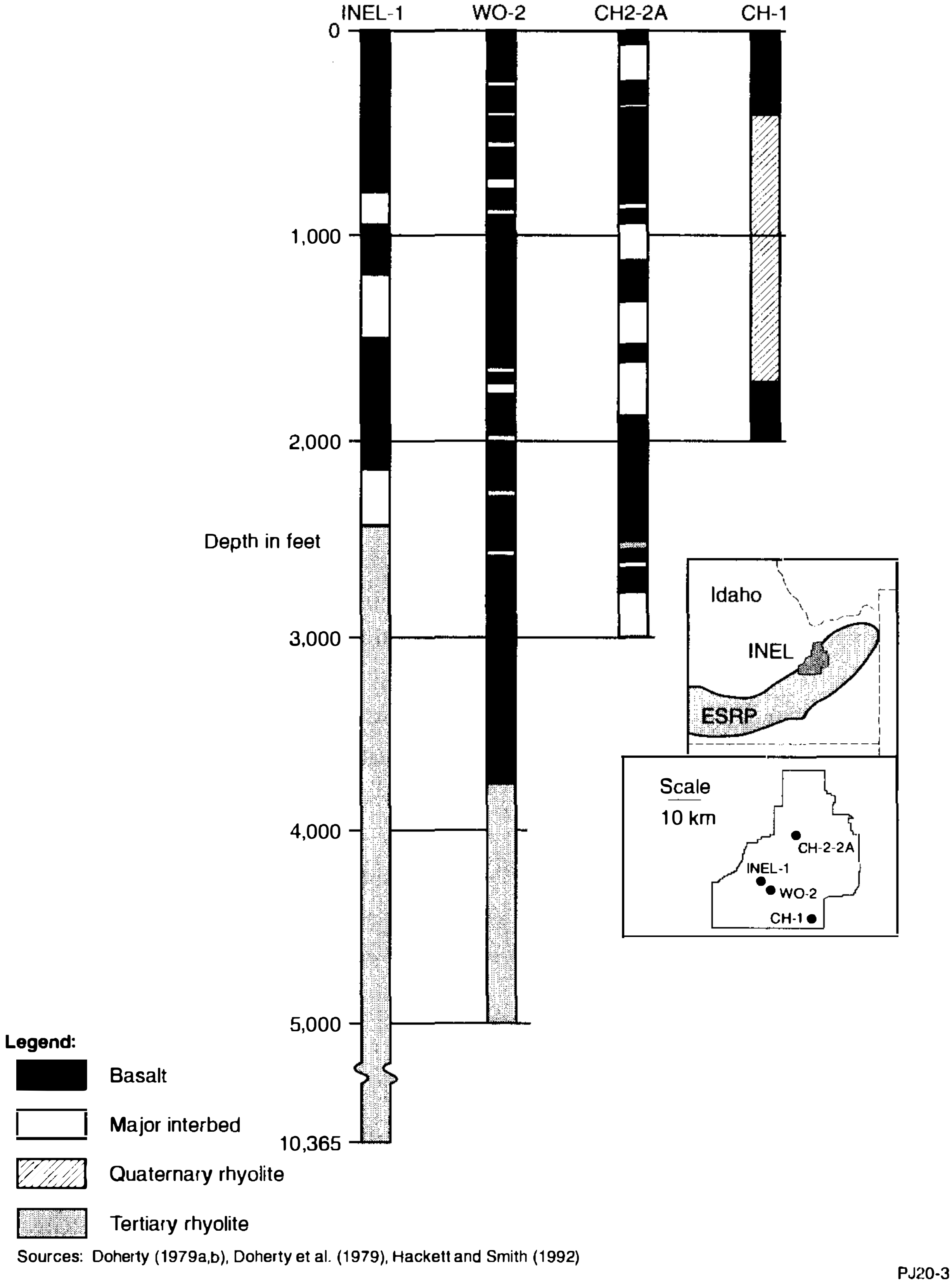

Figure 4.6-2. Lithologic logs of deep drill holes in the INEL area.

4.6.2 Natural Resources

In 1979 the INEL drilled a geothermal exploration well to 3,159 meters (10,365 feet). Researchers measured a temperature of 142°C (288°F) but identified no commercial quantities of geothermal fluids (IDWR 1980). Mineral resources include several quarries or pits inside the INEL boundary that supply sand, gravel, pumice, silt, clay, and aggregate for road construction and maintenance, new facility construction and maintenance, waste burial activities, and ornamental landscaping cinders. During excavations, DOE might study the gravel pits to characterize the local surficial geology of the site. Outside the site boundary, mineral resources include sand, gravel, pumice, phosphate, and base and precious metals (Strowd et al. 1981; Mitchell et al. 1981). The geologic history of the Plain makes the potential for petroleum production at the INEL very low.

4.6.3 Seismic Hazards

The distribution of earthquakes at and near the INEL from 1884 to 1989 clearly shows that the Plain has a remarkably low rate of seismicity, whereas the surrounding Basin and Range has a fairly high rate (Figure 4.6-3, WCC 1992). The mechanism for faulting and generation of earthquakes in the Basin and Range is attributed to northeast-southwest directed crustal extension.

Several investigators have suggested hypotheses for the low rate of seismic activity within the Plain compared to the activity in both the Centennial Tectonic Belt and the Intermountain Seismic Belt:

- Smith and Sbar (1974) and Brott et al. (1981) suggest that high crustal temperatures beneath the Plain and adjacent region inside the seismic parabola (Figure 4.6-1) result in ductile deformation (aseismic creep), in contrast to the brittle deformation (rock fracture) that occurs in the Basin and Range.

- Anders et al. (1989) suggest that the Plain and the adjacent region inside the seismic parabola (Figure 4.6-1) have increased integrated lithospheric strength. They propose that the presence of mid-crustal basic intrusive rock strengthens the crust so that it is too strong to fracture (see also Smith and Arabasz 1991).

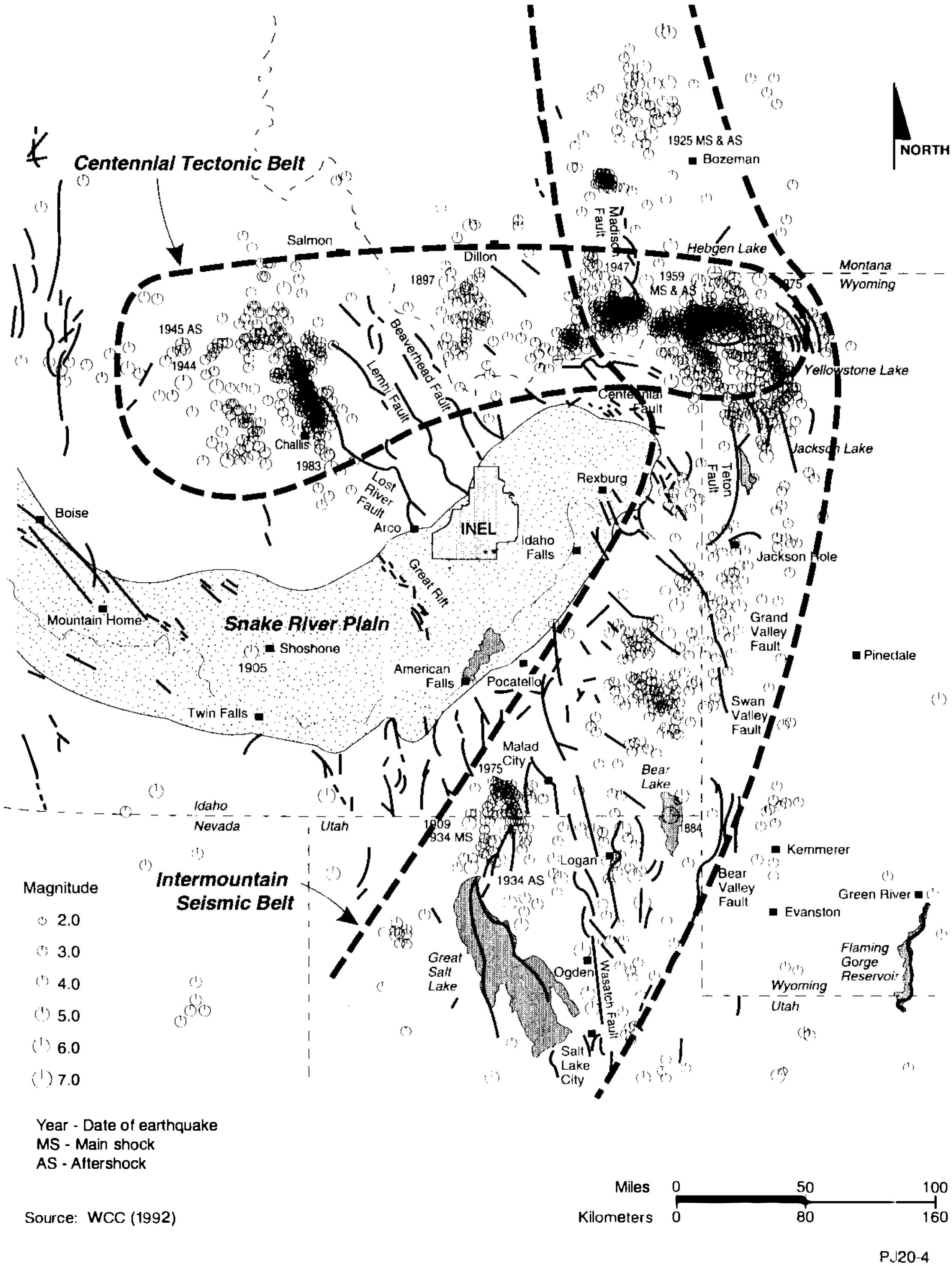

Figure 4.6-3. Earthquakes with magnitudes greater than 2.5 from 1884 to 1989.

- Parsons and Thompson (1991) propose that magma dike injection suppresses normal faulting and associated seismicity by altering the local tectonic stress field. As dikes are injected in volcanic rift zones, they push apart the surrounding rocks and decrease differential stress, thereby preventing earthquakes from occurring.

- Anders and Sleep (1992) propose that the introduction of mantle-derived magma into the midcrust beneath the Plain has decreased faulting and earthquakes by lowering the rate of deformation.

The markedly different tectonic and seismic histories of the Plain and Basin and Range provinces reflect the dissimilar deformational processes acting in each region. Both regions are subjected to the same extensional stress field (Weaver et al. 1979; Zoback and Zoback 1989; Pierce and Morgan 1992; Jackson et al. 1993); however, crustal deformation occurs through dike injection in the Plain and through large-scale normal faulting in the Basin and Range (Rodgers et al. 1990; Parsons and Thompson 1991; Hackett and Smith 1992).

Major seismic hazards include the effects from ground shaking and surface deformation (faulting, tilting). Other potential seismic hazards (e.g., avalanches, landslides, mudslides, soil settlement, and soil liquefaction) are not likely to occur at the INEL because the local geologic conditions are not conducive to them. Based on the seismic history and the geologic conditions, earthquakes greater than moment magnitude 5.5 (and associated strong ground shaking and surface fault rupture) are not likely to occur in the Plain. However, moderate to strong ground shaking from earthquakes in the Basin and Range can affect the INEL. Researchers use patterns of seismicity and locations of mapped faults to assess potential sources of future earthquakes and to estimate levels of ground motion at the site. The sources and maximum magnitudes of earthquakes that could produce the maximum levels of ground motions at all INEL facilities include the following (WCC 1990; WCC 1992):

- A moment magnitude 7.0 earthquake at the southern end of the Lemhi fault along the Howe and Fallert Springs segments

- A moment magnitude 7.0 earthquake at the southern end of the Lost River fault along the Arco segment

- A moment magnitude 5.5 earthquake associated with dike injection in either the Arco or Lava Ridge-Hell's Half Acre Volcanic Rift Zone and the Axial Volcanic Zone

- A "random" moment magnitude 5.5 earthquake occurring in the Eastern Snake River Plain

Figure 4.6-4 shows a facility-specific example of the relationship of the peak ground acceleration on the INEL to the annual frequency of occurrence of seismic events on various seismic sources in the region, including the four events described above (WCFS 1993). The curves refer specifically to the site of the Idaho Chemical Processing Plant in the south-central INEL and might not apply directly to other INEL areas. Ground motion contributions from seismic sources not shown on Figure 4.6-4 (i.e., Intermountain seismic belt and Yellowstone Region) are significantly smaller because of their distant locations or lower estimated maximum magnitudes. The INEL Natural Phenomena Committee determines INEL seismic design-basis events based on studies such as those performed by Woodward Clyde Consultants (1990) and Woodward Clyde Federal Services (1993).

A maximum horizontal ground surface acceleration of 0.24*g* at the Idaho National Engineering Laboratory is estimated to result from an earthquake that could occur once every 2,000 years (DOE 1994). The seismic hazard information presented in this EIS is for general seismic hazard comparisons across DOE sites. Potential seismic hazards for existing and new facilities should be evaluated on a facility-specific basis, consistent with DOE orders, standards, and site-specific procedures. Section 5.15 describes the potential impacts of postulated seismic events.

4.6.4 Volcanic Hazards

Volcanic hazards at the INEL can come from sources inside or outside Plain boundaries. These hazards include the effects of lava flows, ground deformation (fissures, uplift, subsidence), volcanic earthquakes (associated with magmatic processes as distinct from earthquakes associated with tectonics), and ash flows or airborne ash deposits (Bowman 1995). Most of the basalt volcanic activity occurred from 4 million to 2,100 years ago in the INEL area. The most recent and closest volcanic eruption occurred 2,100 years ago at the Craters of the Moon, 25 kilometers (15 miles) southwest of the INEL (Kuntz et al. 1992). The rhyolite domes along the Axial Volcanic Zone formed between 1.2 million and 300,000 years ago and have a recurrence interval of about 200,000 years. Therefore, the probability of future dome formation affecting INEL facilities is very low.

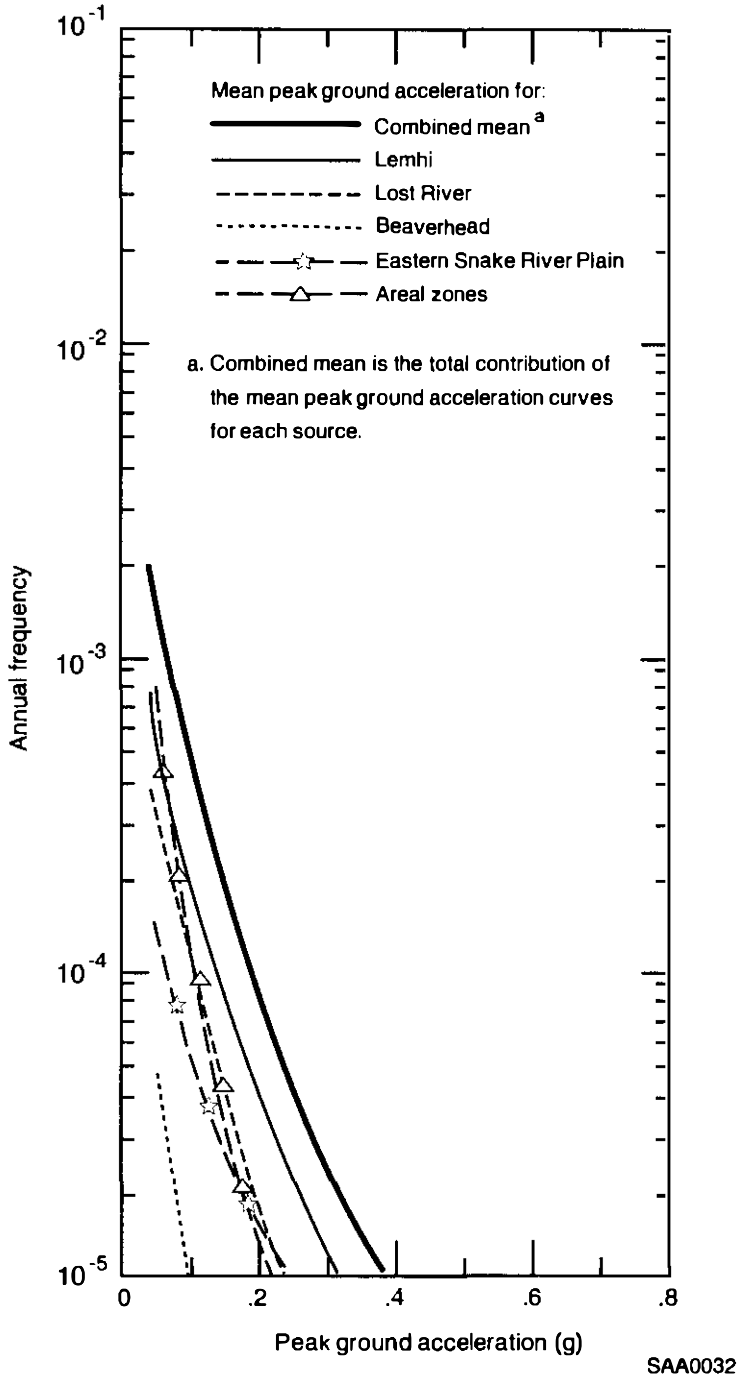

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Figure 4.6-4. Contribution of the seismic sources to the mean peak acceleration at the Idaho Chemical Processing Plant.

Catastrophic Yellowstone eruptions have occurred three times in the past 2 million years, but the INEL is more than 160 kilometers (70 miles) from the Yellowstone Caldera rim and high-altitude winds would not disperse Yellowstone ash in the direction of INEL. Due to the infrequency, great distance, and unfavorable dispersal, pyroclastic flows or ash fallout from future Yellowstone eruptions should not impact the INEL.

Basaltic lava flows and eruptions from fissures or vents might occur. Based on a probability analysis of the volcanic history in the Big Southern Butte area (Volcanism Working Group 1990), the conditional probability that basaltic volcanism would affect a south-central INEL location is less than 2.5×10^{-5} per year (once per 40,000 years or longer), where the risk associated with Axial Volcanic Zone volcanism is greatest. The estimated probability of volcanic impact on INEL facilities farther north, where both silicic and basaltic volcanism have been older and less frequent, is less than 10^{-6} per year (once every million years or longer). The statistics of 116 measured INEL-area lava flow lengths and areas were used to define the two lava flow hazard zones (Figure 4.6-5). The hazard for a particular site within or near a volcanic zone is much lower, typically by an order of magnitude or more, and must be assessed on a site-specific basis (Bowman 1995).

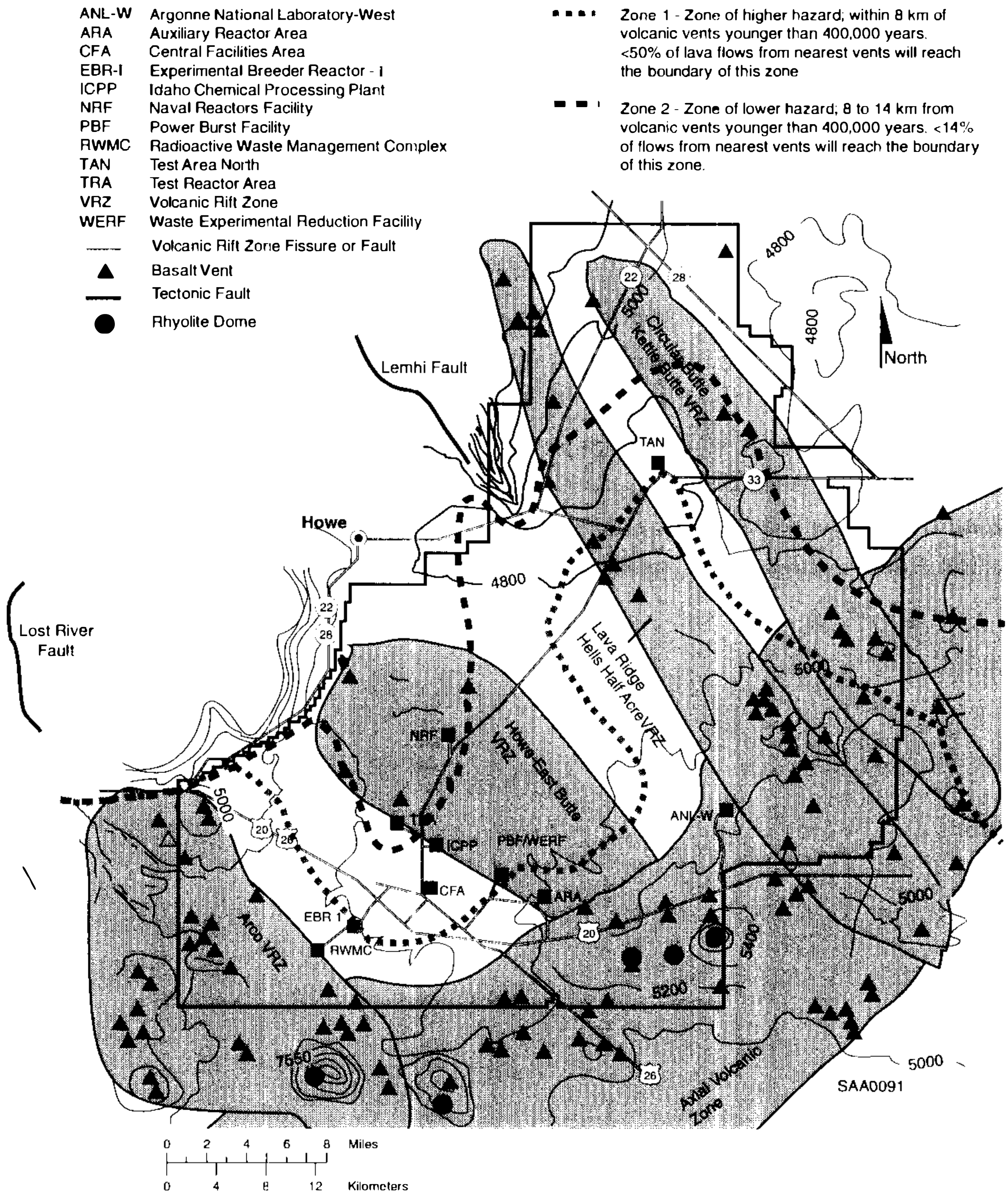

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Figure 4.6-5. Map of the INEL showing locations of volcanic rift zones and lava flow hazard zones.

4.7 Air Quality

This section describes the air resources of the INEL site and the surrounding area. The discussion includes the climatology and meteorology of the region, descriptions of nonradiological and radiological air contaminant emissions, and a characterization of existing and projected levels of air pollutants. The analysis includes both existing facilities and those that were expected (at the time the analysis was performed) to be operational before June 1, 1995. Additional detail and background information on the material presented in this section is presented in Appendix F, Section F-3, of Volume 2.

4.7.1 Climatology and Meteorology

The Eastern Snake River Plain climate exhibits low relative humidity, wide daily temperature swings, and large variations in annual precipitation. Average seasonal temperatures measured on the INEL site range from -7.3°C (18.8°F) in winter to 18.2°C (64.8°F) in summer, with an annual average temperature of about 5.6°C (42°F). Temperature extremes range from a summertime maximum of 39.4°C (103°F) to a wintertime minimum of -45°C (-49°F). The annual average relative humidity is 50 percent, with monthly average maximum values ranging from 59 percent in July to 89 percent in February and December, and with monthly average minimum values ranging from 16 percent in June and July to 47 percent in January (Clawson et al. 1989).

Annual precipitation is light, averaging 221.2 millimeters (8.71 inches), with monthly extremes of zero to 127 millimeters (5 inches). The maximum 24-hour precipitation rate is 46 millimeters (1.8 inches). The greatest short-term precipitation rates are attributable primarily to thunderstorms, which occur approximately two or three days per month during the summer. The average annual snowfall is 701 millimeters (27.6 inches), with a maximum of 1,516 millimeters (59.7 inches) and a minimum of 173 millimeters (6.8 inches) (Clawson et al. 1989).

The INEL site is in the belt of prevailing westerlies; however, the mountain ranges bordering the Eastern Snake River Plain normally channel these winds into a southwest wind. Most offsite locations experience the predominant southwest-northeast wind flow of the Eastern Snake River Plain, although subtle terrain features near some locations cause considerable variations from this flow regime. The annual average wind speed measured at the 6.1-meter (20-foot) level at the Central Facilities Area Weather Station is 3.4 meters per second (7.5 miles per hour). Monthly average values range from

2.3 meters per second (5.1 miles per hour) in December to 4.2 meters per second (9.3 miles per hour) in April and May (Clawson et al. 1989). The highest hourly average near-ground wind speed measured onsite is 22.8 meters per second (51 miles per hour) from the west-southwest, with a maximum instantaneous gust of 34.9 meters per second (78 miles per hour) (Clawson et al. 1989). Figure 4.7-1 presents the frequency of wind speed and wind direction at three meteorological monitoring sites on the INEL site from 1988 to 1992. The wind directions presented in the figure are the direction from which the wind blows. The three wind-roses demonstrate the effects of terrain on predominant wind directions and wind speed. The winds at the Test Area North monitoring station are predominantly from the north-northwest, whereas the winds from the other stations are predominantly from the southwest.

Air pollutant dispersion is a result of the processes of transport and diffusion of airborne contaminants in the atmosphere. Transport is the movement of a pollutant in the wind field, while diffusion refers to the process whereby turbulent eddies dilute a pollutant plume. The temperature gradient of the atmosphere (i.e., the change in temperature with altitude) can restrict or enhance the vertical diffusion of pollutants. Lapse rate conditions, which tend to enhance vertical diffusion, occur slightly less than 50 percent of the time. Conversely, thermal stratification or inversion conditions, which inhibit vertical diffusion, occur slightly more than 50 percent of the time. The height to which the pollutants can freely diffuse is the mixing depth, while the layer of air from the ground to the mixing depth is the mixed layer. Estimates of the monthly average depth of the mixed layer range from 400 meters (1,312 feet) in December to 3,000 meters (9,843 feet) in July. With calm winds and mostly clear skies, nocturnal inversions begin forming after sunset and dissipate about 1 to 2 hours after sunrise. These inversions are often ground-based, meaning the atmospheric temperature increases with height from the ground (Clawson et al. 1989).

Other than thunderstorms, severe weather is uncommon. Five funnel clouds (tornadoes not touching the ground) and no tornadoes were reported on the site between 1950 and 1988. Visibility in the region is good because of the low moisture content of the air and minimal sources of visibility-reducing pollutants. From Craters of the Moon National Monument, the seasonal visual range is from 130 to 155 kilometers (81 to 97 miles) (Notar 1993).

4.7.2 Air Quality

4.7.2.1 Nonradiological Air Quality. The INEL is in the Eastern Idaho Intrastate Air Quality Control Region (AQCR 61). Neither the INEL nor any of the surrounding counties is

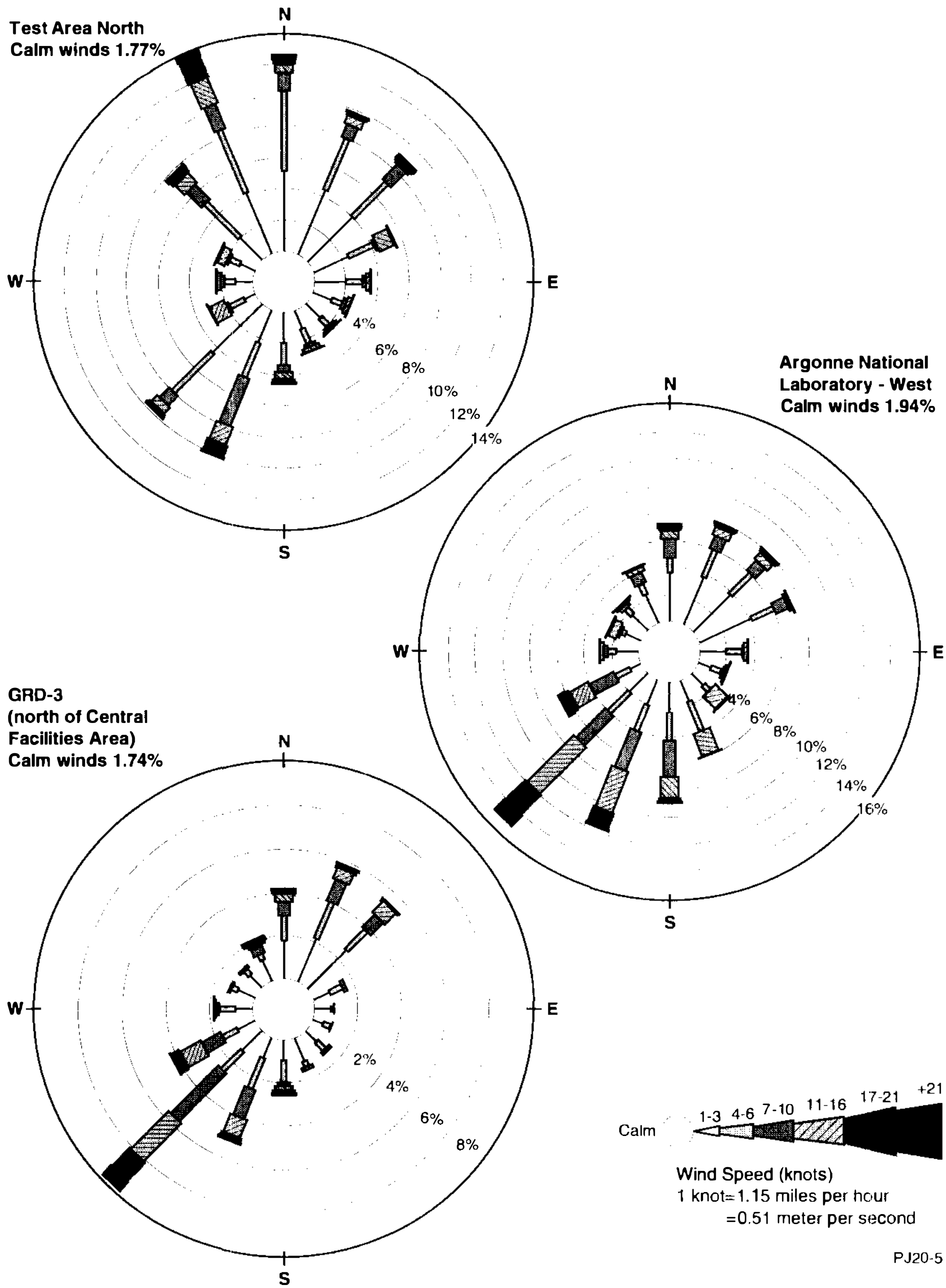

Figure 4.7-1. Depiction of annual average wind direction and speed at INEL meteorological monitoring stations.

designated as a nonattainment area (CFR 1992b) for the National Ambient Air Quality Standards (CFR 1991b). Ambient air quality data monitored in the vicinity of the INEL indicate that the site is in compliance with applicable air quality standards (DOE 1991a).

The Clean Air Act (CAA 1990) contains requirements to prevent the deterioration of air quality in areas designated to be in attainment with the ambient air quality standards. These requirements are administered through a program that limits the increase in specific air pollutants above the levels that existed in what has been termed a baseline (or starting) year, which is 1977. The requirements specify maximum allowable ambient pollutant concentration increases or increments. They specify increment limits for pollutant level increases for the nation as a whole (Class II areas) and prescribe more stringent increment limits (as well as ceilings) for designated national resources, such as national forests, parks, and monuments (Class I areas). Three areas in the INEL vicinity are Prevention of Significant Deterioration Class I ambient air quality areas: Craters of the Moon Wilderness Area, approximately 53 kilometers (33 miles) to the west-southwest; Yellowstone National Park, approximately 143 kilometers (89 miles) to the northeast; and Grand Teton National Park, approximately 145 kilometers (90 miles) to the east-northeast.

DOE evaluates proposed new and modified sources of emissions at INEL to determine the net emissions increase of all pollutants. The INEL is considered a major source, because facility-wide emissions of specific regulated air contaminants exceed 227 metric tons (250 tons) per year. Therefore, a Prevention of Significant Deterioration analysis must be performed for all significant emission increases of specified regulated pollutants. Levels of significance for net emission increases range from very small quantities (less than 1 pound) for beryllium up to 91 metric tons (100 tons) per year for carbon monoxide. Their significance is dependent on the toxicity of the substance. For radionuclides, significance means any increase in emissions that would result in an offsite dose of 0.1 millirem per year or greater.

Ambient air quality standards for Idaho are the same as the National Ambient Air Quality Standards but include total suspended particulates and fluorides. The Idaho Department of Health and Welfare (IDHW) also has ambient concentration limits for hazardous and toxic air pollutants. Table 4.7-1 lists emission rates of criteria and hazardous and toxic air pollutants.

The types and amounts of nonradiological emissions from INEL facilities and activities are similar to those from other industrial complexes that are the same sizes as the INEL. Combustion sources such as boilers and emergency generators emit both criteria and toxic pollutants. Other

Table 4.7-1. Baseline annual average and maximum hourly emission rates of nonradiological air pollutants at the INEL.[a]

Pollutant	Annual average (kg/yr)[b,c]	Maximum hourly (kg/hr)[b]
Criteria pollutants		
Carbon monoxide (CO)	301,000	177
Lead (Pb)	11	0.085
Nitrogen dioxide (NO_2)	744,000	545
Particulate matter (PM_{10})[d]	302,000	230
Sulfur dioxide (SO_2)	202,000	136
Hazardous/toxic air pollutants[e]		
Acetaldehyde	31	0.39
Ammonia	1,600	3.4
Arsenic	4.2	9.0×10^{-4}
Benzene	370	16
1,3-Butadiene	220	0.8
Carbon tetrachloride	28	0.08
Chloroform	1.9	5.5×10^{-3}
Chromium - trivalent	3.1	2.5×10^{-3}
Chromium - hexavalent	0.4	6.2×10^{-4}
Cyclopentane	350	0.58
Dichloromethane	620	0.29
Formaldehyde	960	8.9
Hydrazine	8.3	9.5×10^{-4}
Hydrochloric acid	1,500	0.34
Mercury	200	0.023
Napthalene	16	2.2
Nickel	270	0.057
Nitric acid	1,500	1.7
Phosphorous	56	0.024
Potassium hydroxide	990	0.24
Propionaldehyde	62	0.24
Styrene	4.7	0.74
Tetrachlorethylene	980	0.11
Toluene	580	56
Trichloroethylene	4.7	0.013
Trimethylbenzene	87	12

a. Source: Volume 2, Table 4.7-2.
b. To convert kilograms to pounds, multiply by 2.2.
c. Annual average values include actual emissions plus projected increases from facilities that will become opertional after the baseline year.
d. It is conservatively assumed that all particulate matter is PM_{10} (less than 10 microns in diameter).
e. Hazardous/toxic air pollutants that are listed in State of Idaho regulations and are emitted in levels that exceed screening criteria.

sources include chemical processing operations, transportation, waste management activities, and research laboratories.

Table 4.7-2 compares the INEL contribution to air quality to applicable standards and guidelines. This assessment modelled the INEL air emissions inventory for 1990 using the methodology approved by the U.S. Environmental Protection Agency to predict the maximum ground-level concentration that would occur at or beyond the site boundary for each regulated pollutant (EPA 1993b). The Industrial Source Complex-2 model primarily assessed criteria pollutants, and the SCREEN model assessed toxic air pollutants. The SCREEN model incorporates meteorological data that tend to overestimate impacts, and is useful for identifying cases that require additional, more refined assessments. The baseline concentrations listed in Table 4.7-2 are the sums of the following factors: the concentrations resulting from potential impacts from current operations and the concentrations resulting from the construction or operation of planned upgrades or modifications before the implementation of the proposed actions described in Section 5.7. Background concentrations have not been included because (a) reliable data on background levels in the INEL environs are not available for most pollutants and (b) background levels are low and are more than offset by the use of the maximum (as opposed to actual) baseline. The baseline concentrations represent the maximum calculated concentration occurring at public access locations (site boundary, public roads, and Craters of the Moon Wilderness Area). A comparison of the baseline concentrations to applicable Federal and state criteria pollutant and hazardous/toxic air pollutant guidelines and regulations shows that air quality at INEL is in compliance with those guidelines and regulations. The 24-hour total suspended particulate background concentration is listed as 40 micrograms per cubic meter, which is the same as the annual geometric mean value. The annual sources include chemical processing operations, transportation, waste management activities, and research laboratories.

4.7.2.2 Radiological Air Quality. The major source of radiation exposure in the Eastern Snake River Plain is from natural background radiation sources such as cosmic rays; radioactivity naturally present in soil, rocks, and the human body; and airborne radionuclides of natural origin (such as radon). Sources of radioactivity related to INEL operations include research and training reactors, spent nuclear fuel testing and stabilization, irradiated material and fuel examination, nuclear waste treatment and storage, and depleted uranium armor production.

Radioactive emissions from INEL facilities include the noble gases (argon, krypton, and xenon) and iodine; particulate fission products such as rubidium, strontium, and cesium; radionuclides formed

Table 4.7-2. Comparison of baseline ambient air concentrations with most stringent applicable regulations and guidelines at the INEL.

Pollutant	Averaging time	Most stringent regulation or guideline ($\mu g/m^3$)[a,b,c]	Maximum baseline concentration ($\mu g/m^3$)	Percent of standard
Criteria pollutants				
Carbon monoxide (CO)	8-hour	10,000	280	2.8
	1-hour	40,000	610	1.5
Lead (Pb)	Calendar Quarter	1.5	0.001	<0.1
Nitrogen dioxide (NO_2)	Annual	100	4	4
Particulate matter (PM_{10})	Annual	50	5	10
	24-hour	150	80	53
Sulfur dioxide (SO_2)	Annual	80	6	7.5
	24-hour	365	140	37
	3-hour	1,300	580	45
Hazardous/toxic air pollutants				
Acetaldehyde	Annual	4.5×10^{-1}	1.1×10^{-2}	2
Ammonia	Annual	1.8×10^{2}	6.0×10^{0}	3
Arsenic	Annual	2.3×10^{-4}	9.0×10^{-5}	39
Benzene	Annual	1.2×10^{-1}	2.9×10^{-2}	24
Butadiene	Annual	3.6×10^{-3}	1.0×10^{-3}	28
Carbon Tetrachloride	Annual	6.7×10^{-2}	6.0×10^{-3}	9
Chloroform	Annual	4.3×10^{-2}	4.0×10^{-4}	<1
Chromium - hexavalent	Annual	8.3×10^{-5}	6.0×10^{-5}	72
Chromium - trivalent	Annual	5.0×10^{0}	3.6×10^{-2}	<1
Cylclopentane	Annual	1.7×10^{4}	2.7×10^{-0}	<1
Formaldehyde	Annual	7.7×10^{-2}	1.2×10^{-2}	16
Hydrazine	Annual	3.4×10^{-4}	1.0×10^{-6}	<1
Hydrochloric acid	Annual	7.5×10^{0}	9.8×10^{-1}	13
Mercury	Annual	1.0×10^{0}	4.2×10^{-2}	4
Methylene Chloride	Annual	2.4×10^{-1}	6.0×10^{-3}	3
Napthalene	Annual	5.0×10^{2}	1.8×10^{1}	4
Nickel	Annual	4.2×10^{-3}	2.7×10^{-3}	65
Nitric Acid	Annual	5.0×10^{1}	6.4×10^{-1}	1

Table 4.7-2. (continued).

Pollutant	Averaging time	Most stringent regulation or guideline ($\mu g/m^3$)[a,b,c]	Maximum baseline concentration ($\mu g/m^3$)	Percent of standard
Perchloroethylene	Annual	2.1×10^{0}	1.1×10^{-1}	5
Phosphorous	Annual	1.0×10^{0}	3.0×10^{-1}	30
Potassium hydroxide	Annual	2.0×10^{1}	2.0×10^{-1}	1
Proprionaldehyde	Annual	4.3×10^{0}	3.0×10^{-1}	7
Styrene	Annual	1.0×10^{3}	1.3×10^{0}	<1
Toluene	Annual	3.8×10^{3}	3.7×10^{2}	10
Trichloroethylene	Annual	7.7×10^{-2}	9.7×10^{-4}	1
Trimethylbenzene	Annual	1.2×10^{3}	1.0×10^{2}	8

a. CFR (1991b).
b. IDHW (1994); the ambient standards for the criteria pollutants are the same as the NAAQS.
c. Standards cited for hazardous/toxic air pollutants are for all new sources constructed or modified since May 1, 1994, under State of Idaho Regulations for the Control of Air Pollution in the State of Idaho (IDHW 1994).

Source: Volume 2, Section 4.7.

by neutron activation such as tritium (hydrogen-3), carbon-14, and cobalt-60; and very small quantities (less than 6×10^{-4} curies per year) of heavy elements such as uranium, thorium, plutonium, and their decay products. Historically, the radionuclide with the highest emission rate is the noble gas krypton-85, which is released primarily by the chemical reprocessing of spent nuclear fuel at the Idaho Chemical Processing Plant. Fuel reprocessing also releases small amounts (less than 0.1 curie per year) of iodine-129, which is of concern because of its long half-life (16 million years) and biological properties (iodine isotopes tend to accumulate in the human thyroid). Reactor operations release noble gas isotopes with short half-lives, including argon-41 and isotopes of xenon (primarily xenon-133, -135, and -138). Other activities at the INEL, including waste management operations, result in very low levels of airborne radionuclide emissions (less than 1×10^{-4} curie per year). Table 4.7-3 summarizes airborne radionuclide emissions from INEL facility areas, plus estimated emissions from projects expected, at the time of the analysis was performed, to become operational before June 1, 1995.

Radioactivity released to the atmosphere can result in human exposure through a number of pathways, including inhalation, external exposure, and ingestion. DOE conducts physical

Table 4.7-3. Summary of airborne radionuclide emissions from INEL facility areas (curies per year).[a]

Facility	Tritium/ carbon-14	Iodines	Noble gases	Mixed fission and activation products[b]	U/Th/TRU[c]
Argonne National Laboratory-West	1.0×10^2	—[d]	1.3×10^4	8.1×10^{-4}	1.8×10^{-6}
Central Facilities Area	2.6×10^0	5.0×10^{-7}	—	1.9×10^{-5}	9.6×10^{-7}
Idaho Chemical Processing Plant	4.3×10^1	6.4×10^{-2}	1.0×10^4	3.6×10^{-2}	9.4×10^{-9}
Naval Reactors Facility	1.9×10^{-1}	6.3×10^{-6}	5.7×10^{-1}	5.6×10^{-5}	—
Power Burst Facility/Waste Experimental Reduction Facility	4.9×10^1	—	—	1.3×10^0	9.8×10^{-3}
Radioactive Waste Management Complex	—	—	—	2.6×10^{-5}	4.2×10^{-6}
Test Area North	1.2×10^{-1}	—	—	5.6×10^{-6}	1.5×10^{-5}
Test Reactor Area	1.6×10^2	1.6×10^{-2}	3.3×10^3	3.0×10^0	1.8×10^{-6}
INEL total	2.1×10^3	1.1×10^{-1}	1.2×10^5	5.6×10^0	1.0×10^{-2}

a. With the exception of the Idaho Chemical Processing Plant, emissions estimates are based on 1991 operations. Idaho Chemical Processing Plant emissions are based on 1993 emissions but are scaled upward to reflect operation of the New Waste Calcining Facility at maximum permitted levels. Anticipated projects in the baseline include the Waste Experimental Reduction Facility (compacting and sizing operations but not incineration), Argonne National Laboratory-West Fuel Cycle Facility, and Portable Water Treatment Unit, as described in Appendix F of Volume 2.
b. Mixed fission and activation products that are primarily particulate in nature (for example, cobalt-60, strontium-90, and cesium-137).
c. U/Th/TRU = Radioisotopes of uranium, thorium, or transuranic elements such as plutonium, americium, and neptunium.
d. A dash (—) indicates that the emissions for this group are negligibly small or zero.
Source: Volume 2, Table 4.7-1.

measurements (ambient air monitoring) and uses calculation techniques (atmospheric dispersion modeling) to assess existing levels of radiation (both cosmic and manmade) in and near the site, and to assess doses to workers and the surrounding population.

The offsite population can receive a radiation dose as a result of radiological conditions directly attributable to existing INEL operations. DOE assesses such a dose for a maximally exposed individual and for the population as a whole. The maximally exposed individual is a hypothetical person whose habits and proximity to the site are such that the person would receive the highest dose projected to result from sitewide radioactive emissions. The calculated annual dose to this individual as a result of current and anticipated sitewide emissions is 0.05 millirem (Section 4.7 to Volume 2). This value is a small fraction of both the National Emission Standards for Hazardous Air Pollutants dose limit of 10 millirem per year (CFR 1992a) and the dose received from natural background sources of 351 millirem per year (Section 4.7 to Volume 2). Figure 4.7-2 compares these dose rates.

The collective annual dose to the surrounding population, determined using 1990 U.S. Census Bureau data for the total population residing within an 80-kilometer (50-mile) radius from each facility on the site, is about 0.3 person-rem (Section 4.7 to Volume 2). This value is small in comparison to the annual dose received by the same population from background sources, which is more than 40,000 person-rem (Section 4.7 to Volume 2).

Workers at each major INEL facility can receive radiation exposures. DOE has based its assessment of the dose to these workers on contributions from sources at each facility and those expected to become operational before June 1, 1995. The results of this assessment indicate that the maximum dose received by a worker at any onsite area is about 4.3 millirem per year (Section 4.7 to Volume 2), well below the National Emissions Standard for Hazardous Air Pollutants dose limit of 10 millirem per year. The standard applies to the highest exposed member of the public, and is not applicable to workers. However, it is the most restrictive limit for airborne releases and provides a useful comparison. This dose value of 4.3 millirem per year includes the maximum projected operation of the Portable Water Treatment Unit at the Power Burst Facility Area. However, that operation would be temporary (1 to 2 years) and is not representative of a permanent increase in the baseline. If this facility were not included, the baseline dose to the worker would be about 0.2 millirem per year.

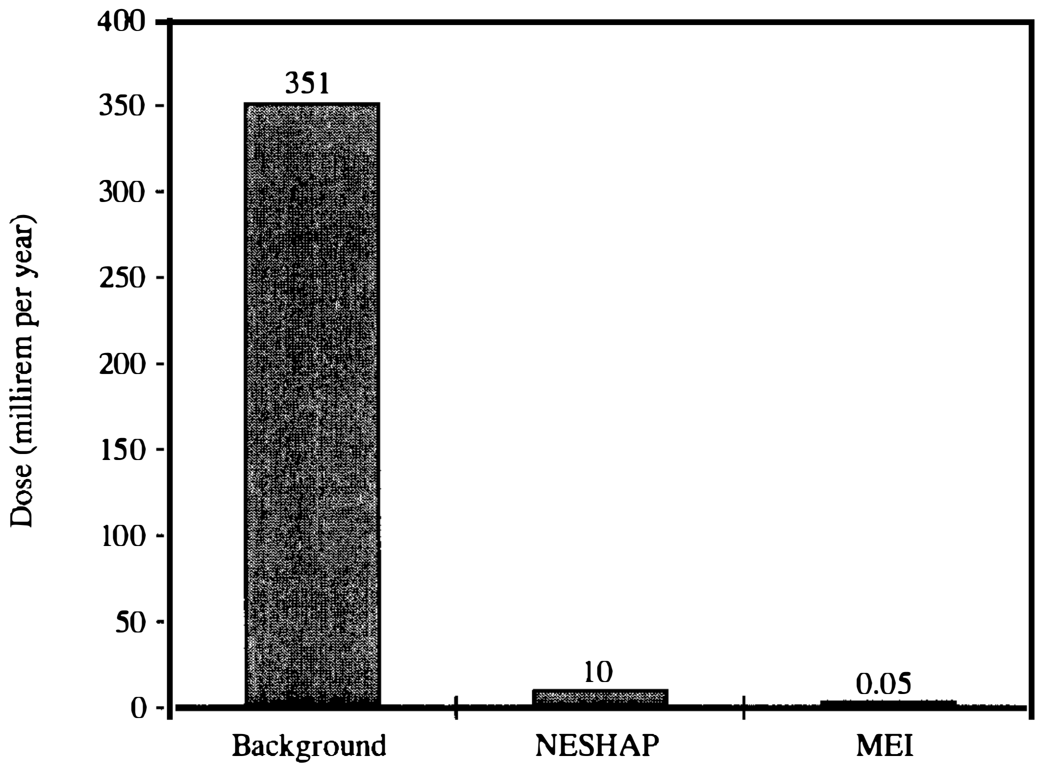

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Figure 4.7-2. Comparison of dose to maximally exposed individual to the National Emission Standard for Hazardous Air Pollutants dose limit and the dose from background sources.

4.8 Water Resources

This section describes existing regional and site hydrologic conditions and discusses the quality of surface and subsurface water and water use and rights. The subsurface water section also describes the vadose zone (or unsaturated zone and perched water bodies) located between the land surface and the water table.

4.8.1 Surface Water

Other than surface-water bodies formed from accumulated runoff during snowmelt or heavy precipitation and manmade infiltration and evaporation ponds, there is little surface water at the site. The following sections discuss regional drainage conditions, local runoff, floodplains, and surface-water quality. Figure 4.8-1 supports discussions in this section.

4.8.1.1 Regional Drainage. The INEL is in the Pioneer Basin, a closed drainage basin that includes three main surface-water bodies--the Big and Little Lost Rivers and Birch Creek. These water bodies drain mountain watersheds directly west and north of the site. However, most of the surface-water flow is diverted for irrigation before it reaches site boundaries (Barraclough et al. 1981), resulting in little or no flow for several years inside the site boundaries (Pittman et al. 1988).

The Big Lost River drains approximately 3,755 square kilometers (1,450 square miles) of land before reaching the site. Approximately 48 kilometers (30 miles) upstream of Arco, Idaho, Mackay Dam controls and regulates the flow of the river, which continues southeast past the towns of Moore and Arco and onto the Eastern Snake River Plain. The river channel then crosses the southwestern boundary of the site, where the INEL Diversion Dam controls surface-water flow. During heavy runoff events, the dam diverts surface water to a series of natural depressions, designated as spreading areas. The Big Lost River continues northeasterly across the site to an area of natural infiltration basins (playas or sinks) near Test Area North. In dry years, surface water does not usually reach the western boundary of the site, and because the INEL is located in a closed drainage basin, surface water never flows off the site.

Birch Creek drains an area of approximately 1,943 square kilometers (750 square miles). In the summer, upstream of the site, surface water from Birch Creek is diverted to provide irrigation and

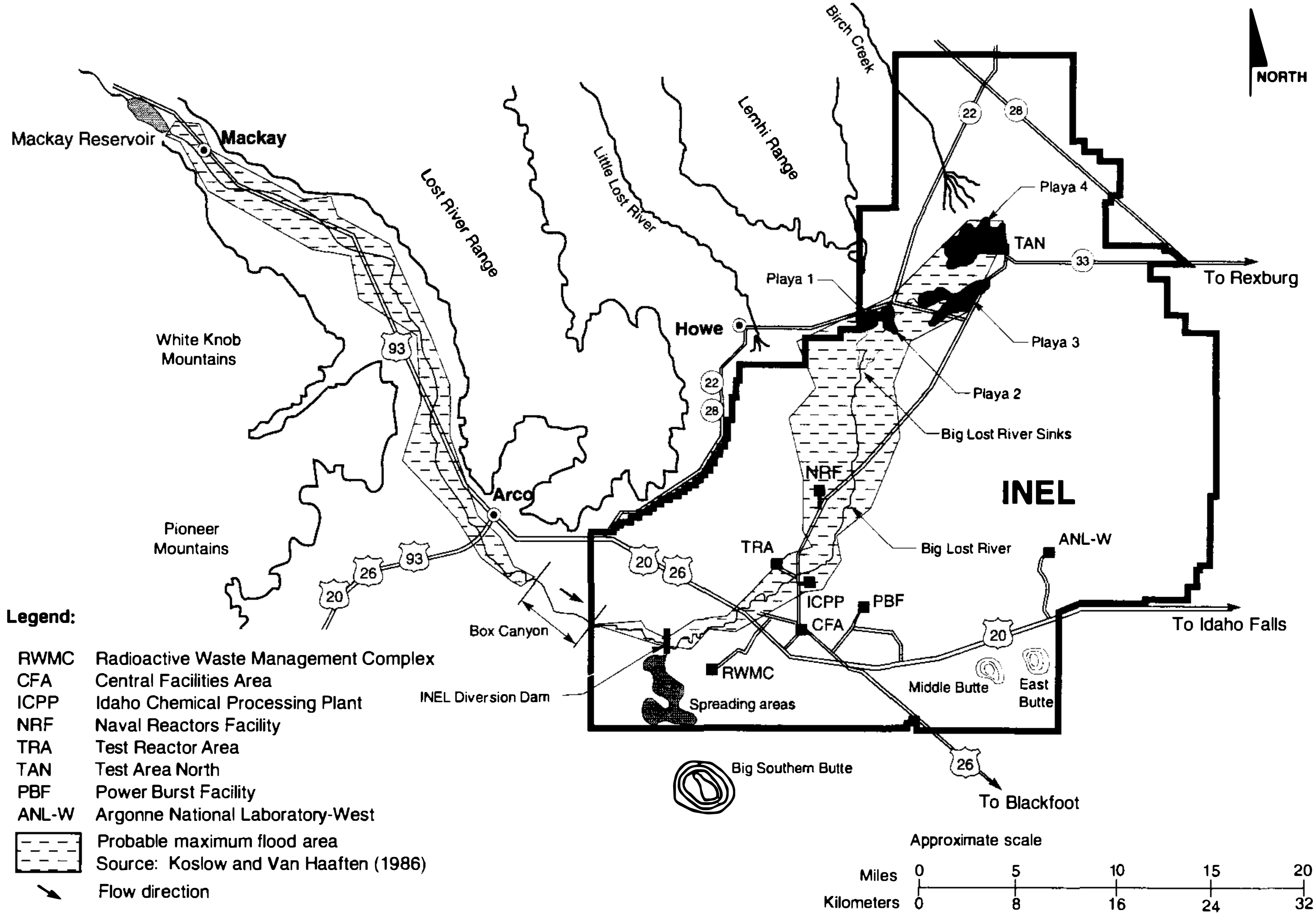


Figure 4.8-1. Selected facilities and predicted inundation map for probable maximum flood-induced overtopping failure of Mackay Dam at the INEL.

to produce hydropower. In the winter, water flow crosses the northwest corner of the site, entering a manmade channel 6.4 kilometers (4 miles) north of Test Area North, where it then infiltrates into channel gravels.

The Little Lost River drains an area of approximately 1,826 square kilometers (705 square miles). Streamflow is diverted for irrigation north of Howe, Idaho. Surface water from the Little Lost River has not reached the site in recent years; however, during high stream flow years, water will reach the site and infiltrate into the subsurface (EG&G 1984).

4.8.1.2 Local Runoff. Surface water generated from local precipitation will flow into topographic depressions (lower elevations than the surrounding terrain) on the site. This surface water either evaporates or infiltrates into the ground, increasing subsurface saturation and enhancing subsurface migration (Wilhelmson et al. 1993).

Localized flooding can occur at the site when the ground is frozen and melting snow combines with heavy spring rains. Test Area North was flooded in 1969 (Koslow and Van Haaften 1986). In 1969 extensive flooding caused by snowmelt occurred in the lower Birch Creek Valley (Koslow 1984). Studies have shown that both the 25- and 100-year, 24-hour rainfall/snowmelt storm event could cause flooding within the Radioactive Waste Management Complex (Dames & Moore 1992). The drainage system, including dikes and erosion prevention features designed to mitigate potential surface water flooding, are being upgraded.

4.8.1.3 Floodplains. Intermittent surface-water flow and the INEL Diversion Dam (built in 1958 and enlarged in 1984) have effectively prevented flooding from the Big Lost River onto the site. However, onsite flooding from the river could occur if high water in the Mackay Dam or the Big Lost River were coupled with a dam failure. Koslow and Van Haaften (1986) examined the consequences of structural failure of the Mackay Dam due to a seismic event, coupled with a probable maximum flood (the largest flood assumed possible in an area). This scenario predicts flood waters overtopping the INEL Diversion Dam and spreading at the Idaho Chemical Processing Plant, Naval Reactors Facility, and the Test Area North Loss-of-Fluid Test Facility (Figure 4.8-1). In the event of a combined Mackay Dam failure and a 100-year flood (flood that occurs on an average of every 100 years), flooding along the Big Lost River would also occur, with low velocities and water depths on the INEL (Koslow and Van Haaften 1986). The area inundated under the Mackay Dam failure scenarios probably would use more than the 100- or 500-year floodplains for the Big Lost River at the INEL. A 100-year floodplain study for the INEL is in progress.

4.8.1.4 Surface-Water Quality. Water quality in the Big and Little Lost Rivers and Birch Creek is similar and has not varied a great deal over the period of record. Measured physical, chemical, and radioactive parameters have not exceeded applicable drinking water quality standards. Chemical composition is determined primarily by the mineral composition of the rocks in the mountain ranges northwest of the site and by the chemical composition of irrigation water in contact with the surface water (Robertson et al. 1974; Bennett 1990).

Site activities do not directly affect the quality of surface water outside the site because discharges from site facilities are to manmade seepage and evaporation basins or stormwater injection wells. Effluents are not discharged to natural surface waters. In addition, surface water does not flow directly off the site (Hoff et al. 1990). However, water from the Big Lost River, as well as seepage from evaporation basins and stormwater injection wells, does infiltrate the Snake River Plain Aquifer (Robertson et al. 1974; Wood and Low 1988; Bennett 1990). These areas are inspected, monitored, and sampled as stipulated in the INEL Stormwater Pollution Prevention Program (DOE-ID 1993b).

4.8.2 Subsurface Water

Subsurface water at the site occurs in the Snake River Plain Aquifer and the vadose zone. This section describes regional and local hydrogeologic conditions, vadose zone hydrology, perched water, and subsurface-water quality. Generally, the term "groundwater" refers to usable quantities of water that enter freely into wells under confined and unconfined conditions within an aquifer (Driscoll 1989).

4.8.2.1 Regional Hydrogeology. The INEL overlies the Snake River Plain Aquifer, the largest aquifer in Idaho (Figure 4.8-2). This aquifer underlies the Eastern Snake River Plain and covers an area of approximately 24,900 square kilometers (9,611 square miles). Groundwater in the aquifer generally flows south and southwestward across the Snake River Plain. The estimated water storage in the aquifer is 2.5×10^{12} cubic meters (2 billion acre-feet, which is about the same as the volume of water contained in Lake Erie) (Robertson et al. 1974). A typical irrigation well can yield as much as 13.9×10^{6} cubic meters (3.7×10^{9} gallons) per year of water if pumped every day (Garabedian 1989). The Snake River Plain Aquifer is among the most productive aquifers in the nation.

The drainage basin recharging the Snake River Plain Aquifer covers an area of approximately 90,643 square kilometers (35,000 square miles). The aquifer is recharged by infiltration of irrigation

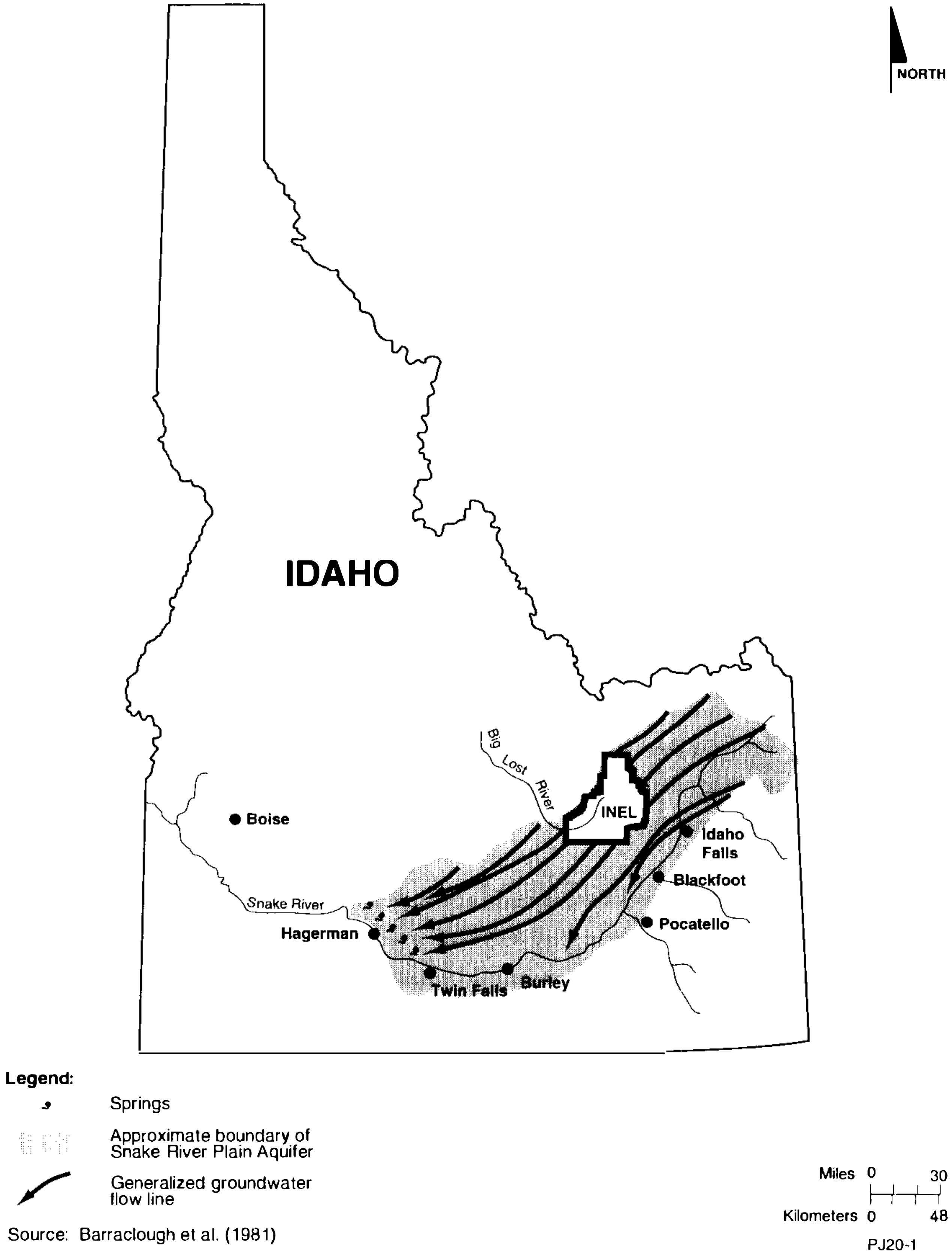


Figure 4.8-2. Location of the INEL, Snake River Plain, and generalized groundwater flow direction of the Snake River Plain Aquifer.

water, seepage from stream channels and canals, underflow from tributary stream valleys extending into the watershed, and direct infiltration from precipitation (Garabedian 1989). Most recharge occurs in surface water-irrigated areas and along the northeastern margins of the plain. Groundwater discharges primarily from the aquifer through springs that flow into the Snake River and from pumping for irrigation. Major springs and seepages that flow from the aquifer are located near the American Falls Reservoir (southwest of Pocatello) and the Thousand Springs area between Milner Dam and King Hill (near Twin Falls).

4.8.2.2 Local Hydrogeology. The INEL site covers 2,305 square kilometers (890 square miles) of the north-central portion of the Snake River Plain Aquifer. Depth to groundwater from the land surface at the site ranges from approximately 61 meters (200 feet) in the north to over 274 meters (900 feet) in the south (Pittman et al. 1988) (see Figure 4.8-3). Groundwater flow is generally toward the south-southwest, and the upper surface is primarily unconfined (not overlain by impermeable soil or bedrock). However, the aquifer behaves as if it were partially confined because of localized geologic conditions. The occurrence and movement of groundwater in the aquifer depends on the geologic setting and the recharge and discharge of water within that setting. Most of the aquifer consists primarily of numerous relatively thin, basaltic lava flows with interbedded sediments extending to depths of 1,067 meters (3,500 feet) below the land surface (Irving 1993). Most of the groundwater migrates horizontally through fractured, basaltic interflow zones (broken and rubble zones) that occur at various depths. Water also migrates vertically along joints and the interfingering edges of interflow zones (Garabedian 1986). Sedimentary interbeds restrict the vertical movement of groundwater. The variability in how the aquifer stores and transmits water increases the difficulty in aquifer investigations and modeling.

The rate at which water moves through the ground depends on the hydraulic gradient (change in elevation and pressure with distance in a given direction) of the aquifer, the effective porosity (percentage of void spaces), and hydraulic conductivity (capacity of a porous media to transport water) of the soil and bedrock. Because aquifer porosity and hydraulic conductivity decrease with depth, most of the water in the aquifer moves through the upper 61 to 152 meters (200 to 500 feet) of the basalts. Estimated flow rates within the aquifer range from 1.5 to 6.1 meters (5 to 20 feet) per day (Barraclough et al. 1981).

The aquifer's ability to transmit water (transmissivity), and its ability to store water (storativity) are important physical properties of the aquifer. In general, the hydraulic characteristics of the aquifer enable the easy transmission of water, particularly in the upper portions.

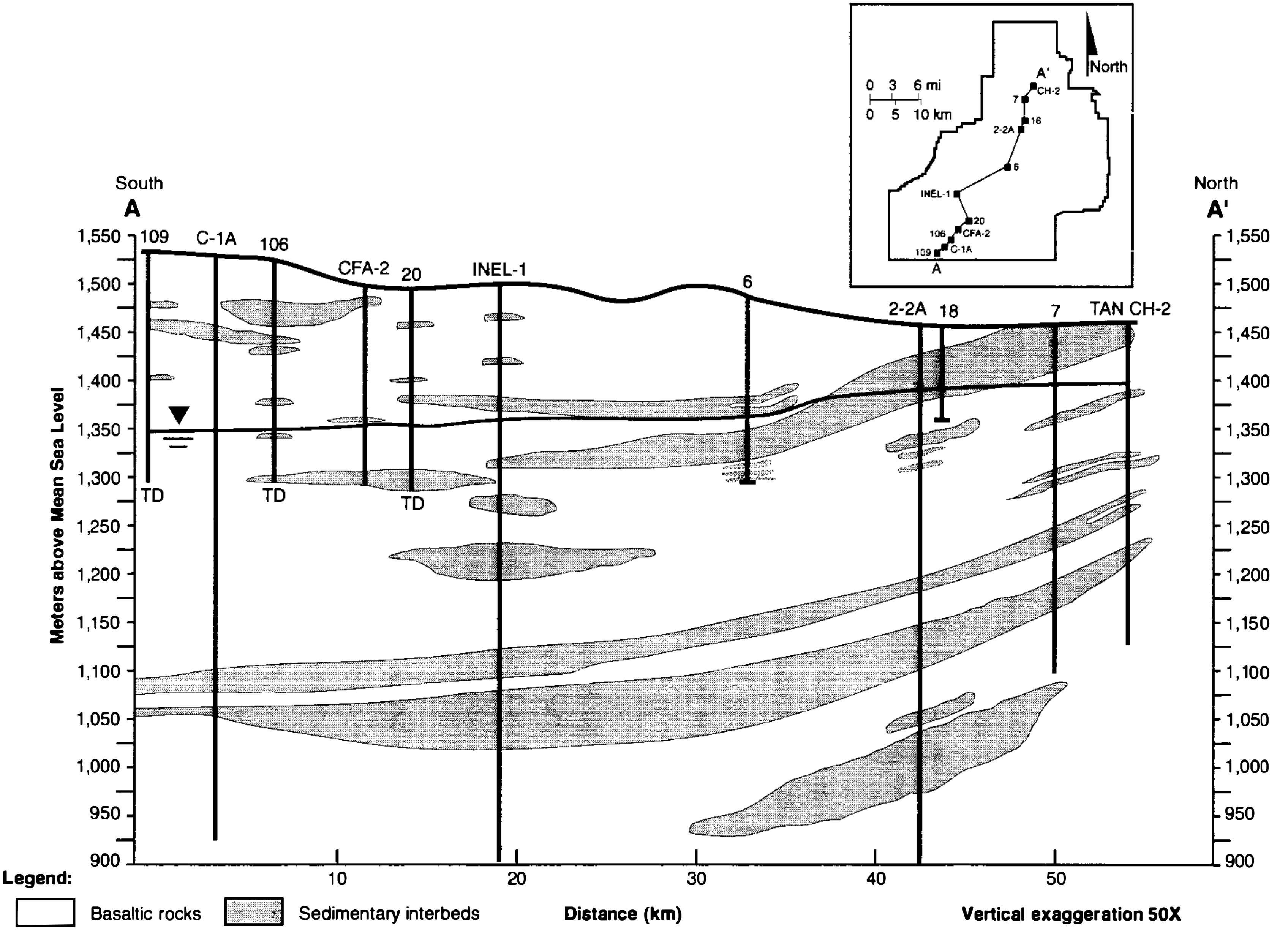


Figure 4.8-3. Hydrostratigraphy across the INEL and water table surface.

Recharge to the aquifer originates off the site from precipitation in the mountains to the west and north. Most of the inflow to the aquifer results from the underflow of groundwater along alluvial-filled valleys adjacent to the Eastern Snake River Plain and adjacent surface-water drainages (i.e., Big and Little Lost Rivers and Birch Creek). In addition, recharge at the site is related to the amount of precipitation, particularly snowfall, for a given year (Barraclough et al. 1981).

4.8.2.3 Vadose Zone Hydrology. The vadose (unsaturated) zone extends from the land surface down to the water table. Within the vadose zone, water and air occupy openings in the geologic materials. Subsurface water in the vadose zone is referred to as vadose water. At the site this complex zone consists of surface sediments (primarily clay and silt, with some sand and gravel) and many relatively thin basaltic lava flows, with some sedimentary interbeds. Thick surficial deposits occur in the northern part of the site, which thin to the south where basalt is exposed at the surface.

The vadose zone protects the groundwater by filtering many contaminants through adsorption, buffering dissolved chemical wastes, and slowing the transport of contaminated liquids to the aquifer. The vadose zone also protects the aquifer by storing large volumes of liquid or dissolved contaminants released to the environment through spills or migration from disposal pits or ponds, allowing natural decay processes to occur.

Travel times for water through the vadose zone are important for an understanding of contaminant movement. The flow rates in the vadose zone depend directly on the extent of fracturing, the percentage of sediments versus basalt, and the moisture content of vadose zone material. Flow increases under wetter conditions and slows under dryer conditions.

4.8.2.4 Perched Water. Locally, saturated conditions that exist above the water table are called perched water. Perched water occurs when water migrates vertically and laterally from the surface until it reaches an impermeable layer (Irving 1993). As perched water spreads laterally, sometimes for hundreds of meters, it moves over the edges of the impermeable layer and continues downward. Several perched water bodies can form between the land surface and the water table.

In general, perched water bodies slow the downward migration of fluids that infiltrate into the vadose zone from the surface because the downward flow is not continuous. The occurrence of perched water at the site is related to the presence of disposal ponds or other surface-water bodies, which studies have detected at the Idaho Chemical Processing Plant, Test Reactor Area, and Test Area North. For example, a 1986 field study at the Idaho Chemical Processing Plant showed that perched water occurs in three areas at possibly three depth zones, ranging from approximately 9 meters (30 feet) to 98 meters (322 feet) below the ground surface and extending laterally as much as 1,097 meters (3,600 feet). In general, the chemical concentrations, shape, and size of these bodies have fluctuated over time in response to the volume of water discharged to the infiltration ponds (Irving 1993).

4.8.2.5 Subsurface Water Quality. Natural water chemistry and contaminants originating at the site affect subsurface water quality. The INEL Groundwater Protection Management Program conducts monitoring programs. This program collects samples from surface water, perched water, and aquifer wells to identify contaminants and contaminant migration to and within the aquifer.

4.8.2.5.1 Natural Water Chemistry — Several factors determine the natural groundwater chemistry of the Snake River Plain Aquifer beneath the site. These factors include the weathering reactions that occur as water interacts with minerals in the aquifer and the chemical composition of (1) groundwater originating outside the site; (2) precipitation falling directly on the land surface; and (3) streams, rivers, and runoff infiltrating the aquifer (Wood and Low 1986, 1988). The chemistry of the groundwater is different, depending on the source areas. For example, groundwater from the northwest contains calcium, magnesium, and bicarbonate leached from sedimentary rocks, and groundwater from the east contains sodium, fluorine, and silicate resulting from contact with volcanic rocks (Robertson et al. 1974).

Although the natural chemical composition of groundwater beneath the site does not exceed the Environmental Protection Agency drinking water standards for any component, the natural chemistry affects the mobility of contaminants introduced into the subsurface from INEL activities. Many dissolved contaminants adsorb (or attach) to the surface of rocks and minerals in the subsurface, thereby retarding the movement of contaminants in the aquifer and inhibiting further migration of contamination. However, many naturally occurring chemicals compete with contaminants for adsorption sites on the rocks and minerals or react with contaminants to reduce their attraction to rock and mineral surfaces.

4.8.2.5.2 Groundwater Quality — Previous waste discharges to unlined ponds and deep wells have introduced radionuclides, nonradioactive metals, inorganic salts, and organic compounds to the subsurface. Table 4.8-1 summarizes the highest detected concentrations of contaminants observed in the aquifer between 1987 and 1992, concentrations near the site boundary, Environmental Protection Agency maximum contaminant levels, and DOE Derived Concentration Guides. The following

Table 4.8-1. Highest detected contaminant concentrations in groundwater at the Idaho National Engineering Laboratory (1987 to 1992).

Parameter	Highest detected recent concentration[a] (year)	Recent boundary condition (year)	Current maximum contaminant level	Derived concentration guide
Radionuclides (picocuries per liter)				
Americium-241	0.91[b] (1990)	< detection limit[c] (1988)	15[d,e]	30[f]
Cesium-137	2,050[b] (1988)	< detection limit[c] (1986)	200[g]	3,000[f]
Cobalt-60	890[b] (1987)	< detection limit[c] (1987)	100[g]	10,000[f]
Iodine-129	3.6[b] (1987)	0.00083-background[h] (1992)	1[g]	500[f]
Plutonium-238	1.28[b] (1990)	< detection limit[c] (1988)	15[d,e]	40[f]
Plutonium-239/240	1.08[b] (1990)	< detection limit[c] (1988)	15[d,e]	30[f]
Strontium-90	640[b] (1992)	< detection limit[c] (1988)	8[g,o]	1,000[f]
Tritium	48,000[b] (1988)	background[i] (1988)	20,000[g]	2,000,000[f]
Nonradioactive metals (milligrams per liter)				
Cadmium	0.0073[b] (1992)	background[c] (1988)	0.005[d]	not applicable
Chromium (total)	0.21[b] (1988)	background[c] (1988)	0.1[d]	not applicable
Lead	0.009[b] (1987)	background[c] (1987)	0.015[g,n]	not applicable
Mercury	0.0004[b] (1987)	background[c] (1987)	0.002[d]	not applicable
Inorganic salts (milligrams per liter)				
Chloride	200[b] (1991)		250[d]	not applicable
Nitrate	5.4[b] (as NO_3) (1988)	background[i] (1988)	10 (as N)[d]	not applicable
Sulfate	140[k] (1985)	background[i] (1985)	250[d]	not applicable
Organic compounds (milligrams per liter)				
Carbon tetrachloride	0.0066[b] (1993)	<detection limit[l] (1988)	0.005[d]	not applicable
Chloroform	0.95[l] (1988)	<detection limit[l] (1988)	0.1[d,m]	not applicable
1,1-dichloroethylene	0.009[b] (1989)	<detection limit[l] (1989)	0.007[d]	not applicable
Cis-1,2-dichloroethylene	3.9[b] (1992)	<detection limit[l] (1988)	0.07[dy]	not applicable
Trans-1,2-dichloroethylene	2.6[b] (1988)	<detection limit[l] (1988)	0.1[d]	not applicable
Tetrachloroethylene	0.051[b] (1992)	<detection limit[l] (1988)	0.005[d]	not applicable
1,1,1-trichloroethane	0.012[b] (1989)	<detection limit[l] (1988)	0.2[d]	not applicable
Trichloroethylene	4.6[b] (1992)	<detection limit[l] (1989)	0.005[d]	not applicable
Vinyl chloride	0.027[l] (1989)	<detection limit[l] (1989)	0.002[d]	not applicable

a. Concentrations are generally for 1987 to 1992.
b. Golder Associates (1994).
c. Orr and Cecil (1991).
d. Maximum contaminant level values taken from EPA (1993a).
e. Maximum contaminant levels have not been established for plutonium-238, plutonium-239, plutonium-240, and americium-241. However, these radionuclides have not been detected above the established limits for gross alpha particle activity (EPA 1993a) or the proposed adjusted gross alpha activity maximum contaminant level for drinking water (FR 1991a).
f. DCGs for radionuclides taken from DOE Order 5400.5, Radiation Protection of the Public and the Environment (DOE 1990b).
g. Maximum contaminant level values taken from (CFR 1991c).
h. Mann (1994).
i. Mann and Cecil (1990).
j. Robertson et al. (1974); Edwards et al. (1990).
k. Pittman et al. (1988).
l. Mann (1990) and Liszewski and Mann (1993).
m. Value is for total trihalomethanes, which is the sum of the concentrations of bromodichloromethane, dibromochloromethane, tribromomethane (bromoform), and trichloromethane (chloroform).
n. Lead action level.
o. Calculated value based on total body or organ dose of 4 millirem per year.

paragraphs discuss each category of contaminants and comparisons of observed concentrations to maximum contaminant levels.

Radionuclides — In general, radionuclide concentrations in the Snake River Plain Aquifer beneath the site have decreased since the mid-1980s because of changes in disposal practices, radioactive decay, adsorption of radionuclides to rocks and minerals, and dilution by natural surface water and groundwater entering the aquifer (Pittman et al. 1988; Orr and Cecil 1991; Bargelt et al. 1992). Radionuclides released and observed in the soil and groundwater include tritium, strontium-90, iodine-129, cobalt-60, cesium-137, plutonium-238, plutonium-239/240, and americium-241 (Golder Associates 1994). Most of these radionuclides have been observed at the Idaho Chemical Processing Plant and Test Reactor Area facility areas. However, radionuclides have also been observed in the Test Area North disposal well.

Concentrations of radionuclides in the aquifer have decreased over time. This decrease is attributed to reduced discharges, adsorption, radioactive decay, and improved waste management practices. As of 1992, concentrations of iodine-129, cobalt-60, tritium, strontium-90, and cesium-137 had exceeded the EPA maximum contaminant levels for radionuclides in drinking water in localized areas inside the INEL boundary. Currently, there are no individual maximum contaminant levels for plutonium-238, plutonium-239, plutonium-240, and americium-241. However, these radionuclides have not been detected above the established limits for gross radioactivity or the proposed adjusted gross alpha activity maximum contaminant level for drinking water (Golder Associates 1994; Mann et al. 1988; Orr and Cecil 1991).

Extremely low concentrations of iodine-129 and tritium have migrated outside site boundaries. In 1992, iodine-129 concentrations were well below the maximum contaminant levels in two wells approximately 6 and 13 kilometers (4 and 8 miles) south of the site boundary (Mann 1994). Tritium concentrations were much below maximum contaminant levels just south of the site boundary in 1985. By 1988 the tritium plume encompassed by the 500 picocurie per liter contour was back inside the site boundary, and its size has continued to decrease (Pittman et al. 1988; Orr and Cecil 1991; Orr et al. 1991). Cobalt-60, strontium-90, cesium-137, plutonium-238, plutonium-240/241, and americium-241 have not been detected outside the site boundaries.

Nonradioactive Metals — The INEL has released sodium, chromium, lead, and mercury on the site and into the subsurface through unlined ponds and deep wells. Of these metals, the INEL released sodium in the greatest quantity from waste treatment processes; however, sodium is not toxic and does not have an established maximum contaminant level. In 1988 chromium concentrations exceeding the maximum contaminant level were measured near the Test Reactor Area. Lead and mercury have occurred at concentrations below the maximum contaminant level near the Idaho Chemical Processing Plant (Orr and Cecil 1991).

Inorganic Salts — Human activities at the site have released chloride, sulfate, and nitrate into the subsurface. Although chloride and sulfate releases have occurred, only nitrate has exceeded maximum contaminant levels (near the Idaho Chemical Processing Plant in 1981). Disposal of nitrates to the injection well and infiltration ponds at the Idaho Chemical Processing Plant account for the elevated nitrate levels in the central portion of the site. By 1988 the levels of nitrate decreased to below the maximum contaminant level. Irrigation in the Mud Lake area might be causing these contaminants to enter the northeastern portion of the site in concentrations comparable to those in nearby irrigated areas (Orr et al. 1991; Robertson et al. 1974; Edwards et al. 1990).

Organic Compounds — Concentrations of volatile organic compounds have been detected in the aquifer beneath the site. However, many of these compounds were detected at amounts below the detection limit (0.002 milligram per liter), or two parts per billion, which is the lowest concentration at which a specific analytical method can detect a contaminant. However, concentrations of the following compounds exceeding the maximum contaminant levels have occurred in and near the Test Area North disposal well: carbon tetrachloride, chloroform, 1,2-cis-dichloroethylene, 1,1-dichloroethylene, 1,2-trans-dichloroethylene, trichloroethylene, tetrachloroethylene, and vinyl chloride (Leenheer and Bagby 1982; Mann and Knobel 1987; Mann 1990; Liszewski and Mann 1992).

4.8.2.5.3 Perched Water Quality — Wastewater discharges from INEL operations have infiltrated into the vadose zone and created most of the perched water beneath the site. Studies have detected elevated concentrations of the following contaminants in samples: tritium, cesium-137, cobalt-60, chromium, and sulfate concentrations in deep perched water near the Test Reactor Area, and strontium-90 in perched water near the Idaho Chemical Processing Plant and at Test Area North (Irving 1993; Schafer-Perini 1993). DOE has not yet measured potential concentrations of contaminants in all INEL perched water bodies. In general, the chemical concentrations, shape, and size of these bodies have fluctuated over time in response to the volume of water discharged to the infiltration ponds.

4.8.3 Water Use and Rights

The INEL does not withdraw or use surface water for site operations, nor does it discharge effluents to natural surface water. However, the three surface-water bodies at or near the site (Big and Little Lost Rivers and Birch Creek) have the following designated uses: agricultural water supply, cold-water biota, salmonid spawning, and primary and secondary contact recreation. In addition, waters in the Big Lost River and Birch Creek have been designated for domestic water supply and as special resource waters.

Groundwater use on the Snake River Plain includes irrigation, food processing and aquaculture, and domestic, rural, public, and livestock supply. Water use for the upper Snake River drainage basin and the Snake River Plain Aquifer was 16.4 billion cubic meters (4.3 trillion gallons) per year in 1985, which was more than 50 percent of the water used in Idaho and approximately 7 percent of agricultural withdrawals in the nation. Most of the water withdrawn from the Eastern Snake River Plain [1.8 billion cubic meters (0.47 trillion gallons) per year] is for agriculture. The aquifer is the source of all water used at the INEL. Site activities withdraw water at an average rate of 7.4 million cubic meters (1.9 billion gallons) per year (DOE-ID 1993e). However, the baseline annual withdrawal rate dropped to 6.5 million cubic meters (1.7 billion gallons) in 1995. The average annual withdrawal is equal to approximately 0.4 percent of the water consumed from the Eastern Snake River Plain Aquifer, or 53 percent of the maximum annual yield of a typical irrigation well. Of the quantity of water pumped from the aquifer, a substantial portion is discharged to the surface or subsurface and eventually returned to it (DOE-ID 1993d,e).

A sole-source aquifer, as designated by the Safe Drinking Water Act (SDWA 1974) is one that supplies 50 percent of the drinking water consumed in the area overlying the aquifer. Sole-source aquifer areas have no alternative source or combination of sources that could physically, legally, and economically supply all those who obtain their drinking water from the aquifer. Because groundwater supplies 100 percent of the drinking water consumed within the Eastern Snake River Plain (Gaia Northwest 1988) and an alternative drinking water source or combination of sources is not available, the Environmental Protection Agency designated the Snake River Plain Aquifer a sole-source aquifer in 1991 (FR 1991b).

DOE holds a Federal Reserved Water Right for the INEL, which permits a water pumping capacity of 2.3 cubic meters (80 cubic feet) per second and a maximum water consumption of 43 million cubic meters (11.4 billion gallons) per year for drinking, process water, and noncontact cooling. Because it is a Federal Water Right, the site's priority on water rights dates back to the establishment of the INEL.

4.9 Ecological Resources

This section describes the biotic resources — flora, fauna, threatened and endangered species, and wetlands — on the INEL site, which are typical of the Great Basin and Columbia Plateau. Because the proposed actions are most likely to affect areas near existing major facilities, this section emphasizes the biotic resources in those areas. However, because the proposed actions could affect other resources outside such areas (e.g., more mobile species like pronghorn, *Antilocapra americana*), it also describes biotic resources for the entire INEL site.

4.9.1 Flora

Vegetation on the INEL site is primarily of the shrub-steppe type and is a small fraction of the 45,000 square kilometers (111.2 million acres) of this vegetation type in the Intermountain West. The 15 vegetation associations on the INEL site range from primarily shadscale-steppe vegetation at lower altitudes through sagebrush- and grass-dominated communities to juniper woodlands along the foothills of the nearby mountains and buttes (Rope et al. 1993; Kramber et al. 1992; Anderson 1991). These associations can be grouped into six basic types: juniper woodland, grassland, shrub-steppe (which consists of "sagebrush-steppe" and "salt desert shrubs"), lava, bareground-disturbed, and wetland vegetation. Shrub-steppe vegetation, which is dominated by big sagebrush (*Artemisia tridentata*), saltbush (*Atriplex* spp.), and rabbitbrush (*Chrysothamnus* spp.) covers more than 90 percent of the INEL. Grasses include cheatgrass (*Bromus tectorum*), Indian ricegrass (*Oryzopsis hymenoides*), wheatgrasses, (*Agropyron* spp.), and squirreltail (*Sitanion hysterix*). Herbaceous plants include phlox (*Phlox* spp.), wild onion (*Allium* spp.), milkvetch (*Astragalus* spp.), Russian thistle (*Salsola kali*), and various mustards. Work being conducted by Idaho State University will provide additional information on INEL plant communities and the status of sensitive plant species.

Facility and human-disturbed (grazing not included) areas cover only about 2 percent of the INEL. Introduced annuals, including Russian thistle and cheatgrass, frequently dominate disturbed areas. These species usually are less desirable to wildlife as food and cover, and compete with more desirable perennial native species. These disturbed areas serve as a seed source, increasing the potential for the establishment of Russian thistle and cheatgrass in surrounding less-disturbed areas. Vegetation inside facility boundaries is generally disturbed or landscaped. Species richness on the INEL is comparable to that of like-sized areas with similar terrain in other parts of the Intermountain West. Plant diversity is typically lower in disturbed and modified areas.

4.9.2 Fauna

The INEL site supports animal communities characteristic of shrub-steppe vegetation and habitats. More than 270 vertebrate species occur, including 46 mammal, 204 bird, 10 reptile, 2 amphibian, and 9 fish species (Arthur et al. 1984; Reynolds et al. 1986). Common small-mammal genera include mice (*Reithrodontomys* spp. and *Peromyscus* spp.), chipmunks (*Tamias* spp.), jackrabbits (*Lepus* spp.), and cottontails (*Sylvilagus* spp.).

Songbirds and passerines commonly observed at the INEL include the American robin (*Turdus migratorius*), horned lark (*Eremophila alpestris*), black-billed magpie (*Pica pica*), sage thrasher (*Oreoscoptes montanus*), Brewer's sparrow (*Spizella breweri*), sage sparrow (*S. belli*), and western meadowlark (*Sturnella neglecta*), while resident upland gamebirds include the sage grouse (*Centrocercus urophasianus*), chukar (*Alectoris chukar*), and grey partridge (*Perdix perdix*). Common migratory bird species, which use the INEL for part of the year, include a variety of waterfowl [e.g., mallard (*Anas platyrhynchos*), northern pintail (*Anas acuta*), and Canada goose (*Branta canadensis*)] and raptors [e.g., Swainson's hawk (*Buteo swainsoni*), rough-legged hawk (*B. lagopus*), and American kestrel (*Falco sparverius*)].

The most abundant big-game species that occurs on the INEL is the pronghorn, but mule deer (*Odocoileus hermonius*), moose (*Alces alces*), and elk (*Cervus elaphus*) are present in small numbers as transients. Other large mammals observed on the INEL include the coyote (*Canis latrans*), which is common across the site, and the badger (*Taxidea taxus*) and bobcat (*Felis rufus*), both of which are present across the site but are much less abundant. Fish, including kokanee salmon (*Oncorhynchos nerka*), rainbow trout (*Oncorhynchos mykiss*), and mountain whitefish (*Prosopium williamsoni*), occur on the INEL only when the Big Lost River flows onto the site (as a result of heavy rain- or snowfall in the mountains to the northwest); they are not full-time residents.

A number of researchers have studied effects of radiation exposure from contaminated areas at INEL on small mammals and birds, and have concluded that subtle sublethal effects (e.g., reduced growth rates and life expectancies) can occur in individual animals as a result of radiation exposure. However, they can attribute no population or community-level impacts to such exposures (Halford and Markham 1978; Evenson 1981; Arthur et al. 1986; Millard et. al 1990).

The monitoring of radionuclide levels outside the boundaries of the various INEL facilities and off the INEL site has detected radionuclide concentrations above background levels in individual plants and animals (Markham 1974; Craig et al. 1979; Markham et al. 1982; Morris 1993), but these limited data suggest that populations of exposed animals (e.g., mice and rabbits) as well as animals that feed on these exposed animals (e.g., eagles and hawks) are not at risk.

4.9.3 Threatened, Endangered, and Sensitive Species

State and Federal regulatory agency lists (Lobdell 1992, 1995), the Idaho Department of Fish and Game Conservation Data Center list, and information from site surveys provided the information to identify Federal- and state-protected, candidate, and sensitive species that potentially occur on the INEL. This information identified two Federal endangered (bald eagle, and peregrine falcon) and nine Federal Category 2 candidate (white-faced ibis, northern goshawk, ferruginous hawk, burrowing owl, long-eared myotis, small-footed myotis, pygmy rabbit, Townsend's western big-eared bat, and Idaho pointheaded grasshopper) species as animals that potentially occur on the INEL site (Table 4.9-1). Five animal species listed by the state as Species of Special Concern occur on the site. No frequent observations of the Federal- or state-listed animal species have occurred near any of the facilities where proposed actions would occur. This analysis did not identify any Federal- or state-listed plant species as potentially occurring on the INEL site. Eight plant species identified by other Federal agencies and the Idaho Native Plant Society as sensitive, rare, or unique occur on the site (Chowlewa and Henderson 1984).

4.9.4 Wetlands

The U.S. Fish and Wildlife Service National Wetlands Inventory has identified more than 130 areas inside the boundaries of the INEL that might possess some wetlands characteristics. Surveys conducted in the fall of 1992 indicate that these possible wetlands cover about 1.4 percent (33 square kilometers or 8,206 acres) of the INEL site (Hampton et al. 1993). Approximately 70 percent of these possible wetlands areas occur near the Big Lost River and its spreading areas and playas, near the Birch Creek Playa, and in an area north of and in the general vicinity of Argonne National Laboratory-West. Limited riparian (riverbank) communities with mature trees along the Big Lost River (Reynolds 1993) reflect the intermittent flow in the river (1986 and 1993 were the last two years with flow reported on the site). The remainder of the possible wetlands are scattered throughout the INEL site. In 1994, INEL began evaluating these potential wetlands to determine if they meet the Corps of Engineers definition of jurisdictional wetlands (COE 1987). Approximately 20 wetlands are near facilities and are mostly manmade (e.g., industrial waste and sewage treatment ponds, borrow pits, and gravel pits).

Table 4.9-1. Threatened and endangered species, special species of concern, and sensitive species that may be found on the INEL.

	Name	Status[a]	Comments
BIRDS	Northern goshawk *(Accipiter gentilis)*	C2, SSC, FS, BLM	The **ferruginous hawk** nests on and migrates through the INEL. This species is found throughout the INEL but is observed more frequently in juniper woodlands. The **peregrine falcon** has been observed rarely in winter, but has not been observed during other seasons. The last sighting was in 1993 (Morris 1993). It is not known to nest on the INEL and is not commonly observed near facilities (Reynolds 1993a). The **bald eagle** is a winter resident and is locally common in the far north end and on the western edge of the INEL near Howe (Reynolds 1993a). It is not known to nest on the INEL and is not commonly observed near facilities (Reynolds 1993). The **white-faced ibis**, which uses aquatic and riparian habitats, is an uncommon migrant at the INEL. The **long-billed curlew** is known to nest on the north end of the INEL near agricultural lands. The **northern goshawk** is a casual migrant through the INEL.
	Burrowing owl *(Athene cunicularia)*	C2, BLM	
	Ferruginous hawk *(Buteo regalis)*	C2, SSC, BLM	
	Swainson's hawk *(Buteo swainsoni)*	BLM	
	Great egret *(Casmerodius albus)*	SSC	
	Merlin *(Falco columbarius)*	SSC, BLM	
	Peregrine falcon *(Falco peregrinus)*	E	
	Gyrfalcon *(Falco rusticolus)*	BLM	
	Common loon *(Gavia immer)*	SSC, FS	
	Bald eagle *(Haliaeetus leucocephalus)*	E	
	Long-billed curlew *(Numenius americanus)*	SPS, BLM	
	American white pelican *(Pelecanus erythrorhynchos)*	SSC	
	White-faced ibis *(Plegadis chihi)*	C2	
MAMMALS	Merriam's shrew (*Sorex merriami*)	SPS	The **pygmy rabbit** is common on the INEL, but its distribution is patchy (Reynolds et al. 1986). Roosts and hibernation caves for **Townsend's western big-eared bat** occur on the INEL. All are over 7 kilometers (3 miles) from facilities. Brood caves might exist on the site but have not been located.
	Pygmy rabbit *(Brachylagus (Sylvilagus) idahoensis)*	C2, BLM, SSC	
	California myotis *(Myotis californicus)*	SSC	
	Fringed myotis *(Myotis thysanodes)*	SSC	
	Western pipistrelle *(Pipistrellus hesperus)*	SSC, BLM	
	Townsend's western big-eared bat *(Plecotus townsendii)*	C2, SSC, FS, BLM	
	Long-eared myotis (*Myotis evotis*)	C2	
	Small-footed myotis (*Myotis subulatus*)	CS	
PLANTS	Lemhi milkvetch *(Astragalus aquilonius)*	BLM, FS, INPS	The 8 plant species identified as sensitive, rare, or unique that are known to occur on the INEL occur primarily at a distance from INEL facilities and are uncommon on the INEL because they require unique microhabitat conditions.
	Painted milkvetch *(Astragalus ceramicus var. apus)*	3c, INPS-M	
	Winged-seed evening primrose *(Camissonia pterosperma)*	BLM, INPS-S	
	Nipple cactus *(Coryphantha missouriensis)*	INPS-M	
	Spreading gilia *(Ipomopsis (Gilia) polycladon)*	BLM, INPS-2	
	King's bladderpod (*Lesquerella kingii var. cobrensis*)	INPS-M	
	Tree-like oxytheca *(Oxytheca dendroidea)*	INPS-S	
	Sepal-tooth dodder (*Cuscuta denticulata*)	INPS-1	
INSECTS	Idaho pointheaded grasshopper (*Acrolophitus pulchellus*)	C2, BLM	Occurs just north of the INEL.

a. Key:
- C2 = Federal Category 2 species.
- 3c = No longer considered for Federal listing.
- E = Federal and state endangered species.
- SSC = State species of special concern.
- BLM = Bureau of Land Management monitored.
- FS = U.S. Forest Service monitored.
- INEL = Idaho National Engineering Laboratory.
- SPS = State protected species.
- INPS-S = Idaho Native Plant Society sensitive.
- INPS-M = Idaho Native Plant.
- INPS-1 = Idaho Native Plant Society State Priority 1.
- INPS-2 = Idaho Native Plant Society State Priority 2.

4.10 Noise

The major noise sources at the INEL occur primarily in developed operational areas. These sources include facilities; equipment and machines (e.g., cooling towers, transformers, engines, pumps, boilers, steam vents, paging systems, construction equipment, and materials-handling equipment); aircraft; and bus, car, truck, and railroad traffic. At the INEL boundary, which is more than 3 kilometers (2 miles) from any facility, noise from most sources is barely distinguishable from background noise levels. Some disturbance of wildlife activities could occur at the INEL as a result of noise from operational and construction activities. The State of Idaho and the counties in which the INEL is located have not established any regulations that specify acceptable community noise levels, with the exception of prohibitions on nuisance noise.

Existing INEL-related noises of public significance are from the transportation of people and materials to and from the site and in-town facilities via buses, trucks, private vehicles, helicopters, and freight trains. During the normal workweek, most of the 4,000 to 5,000 employees who work on the site (as opposed to those working in Idaho Falls) travel daily by buses from surrounding communities (see Section 4.3). In addition, 300 to 500 private vehicles travel to the INEL site from surrounding communities each day (see Section 4.11). Noise measurements along U.S. Highway 20 about 15 meters (50 feet) from the roadway indicate that the sound level from traffic ranges from 64 to 86 decibels, A-weighted (dBA) (Abbott et al. 1990), and that the primary source is buses (71 to 81 dBA). While few people reside within 15 meters (50 feet) of the roadway, the results indicate that INEL traffic noise might be objectionable to members of the public residing near principal highways or busy bus routes. The acoustic environment along the INEL site boundary in rural areas and at nearby areas away from traffic noise is typical of a rural location, with the day-night sound level (DNL) in the range of 35 to 50 dBA (EPA 1974).

Public exposure to aircraft noise is due in part to INEL-related activities. Air cargo and business travel of INEL personnel via commercial air transport is a significant fraction of all such travel in and out of regional airports. Onsite INEL security patrol and surveillance flights do not adversely affect individuals off the site because of the INEL's remoteness. For INEL helicopter flights that originate or terminate in Idaho Falls, members of the public are exposed to the unique noises produced by these aircraft. Because the number of flights per day is limited and most flights occur during nonsleeping hours, public exposure to aircraft nuisance noise is not great.

Normally only one train per day serves the INEL, via the Scoville spur. Noise sources related to rail transport include those from diesel engines, wheel-track contact, and whistle warnings at rail crossings. Even with only one or two exposures to these sources per day, individuals residing near the railroad tracks might find the noises mildly objectionable.

4.11 Traffic and Transportation

Roads are the primary access to and from the INEL site. Commercial shipments are transported via truck and plane, some bulk materials are transported via rail, and waste is transported by road and rail. This section discusses the existing traffic volumes, transportation routes, transportation accidents, and waste and materials transportation, including baseline radiological exposures from waste and materials transportation. This section summarizes the information in Lehto (1993).

4.11.1 Roadways

4.11.1.1 Infrastructure Regional and Site Systems. Figure 4.11-1 shows the existing regional highway system. Two interstate highways serve the regional area. Interstate 15 (I-15), a north-south route that connects several cities along the Snake River, is approximately 40 kilometers (25 miles) east of the INEL site. I-86 intersects I-15 approximately 64 kilometers (40 miles) south of the INEL site, and provides a primary linkage from I-15 to points west. I-15 and US 91 are the primary access routes to the Shoshone-Bannock reservation. US 20 and US 26 are the main access routes to the southern portion of the INEL site. Idaho State Routes 22, 28, and 33 pass through the northern portion of the INEL; State Route 33 provides access to the northern INEL site facilities. Table 4.11-1 lists the baseline (1991) traffic for several of these access routes. The level of service of these segments is currently designated "free flow," which is defined as "operation of vehicles is virtually unaffected by the presence of other vehicles."

The INEL has developed an onsite road system of approximately 140 kilometers (87 miles) of paved surface, including about 29 kilometers (18 miles) of service roads that are closed to the public. Most of the roads are adequate for the current level of normal transportation activity and could handle some increased traffic volume. DOE plans to reconstruct several deteriorating INEL roads built in the 1950s that have been and will continue to be used to transport heavier-than-normal loads.

4.11.1.2 Infrastructure Idaho Falls. Approximately 4,000 DOE and contractor personnel administer and support INEL work at offices in Idaho Falls. DOE shuttle vans provide hourly transport between in-town facilities. One of the busiest intersections is Science Center Drive and Fremont Avenue, which serves Willow Creek Building, Engineering Research Office Building, INEL

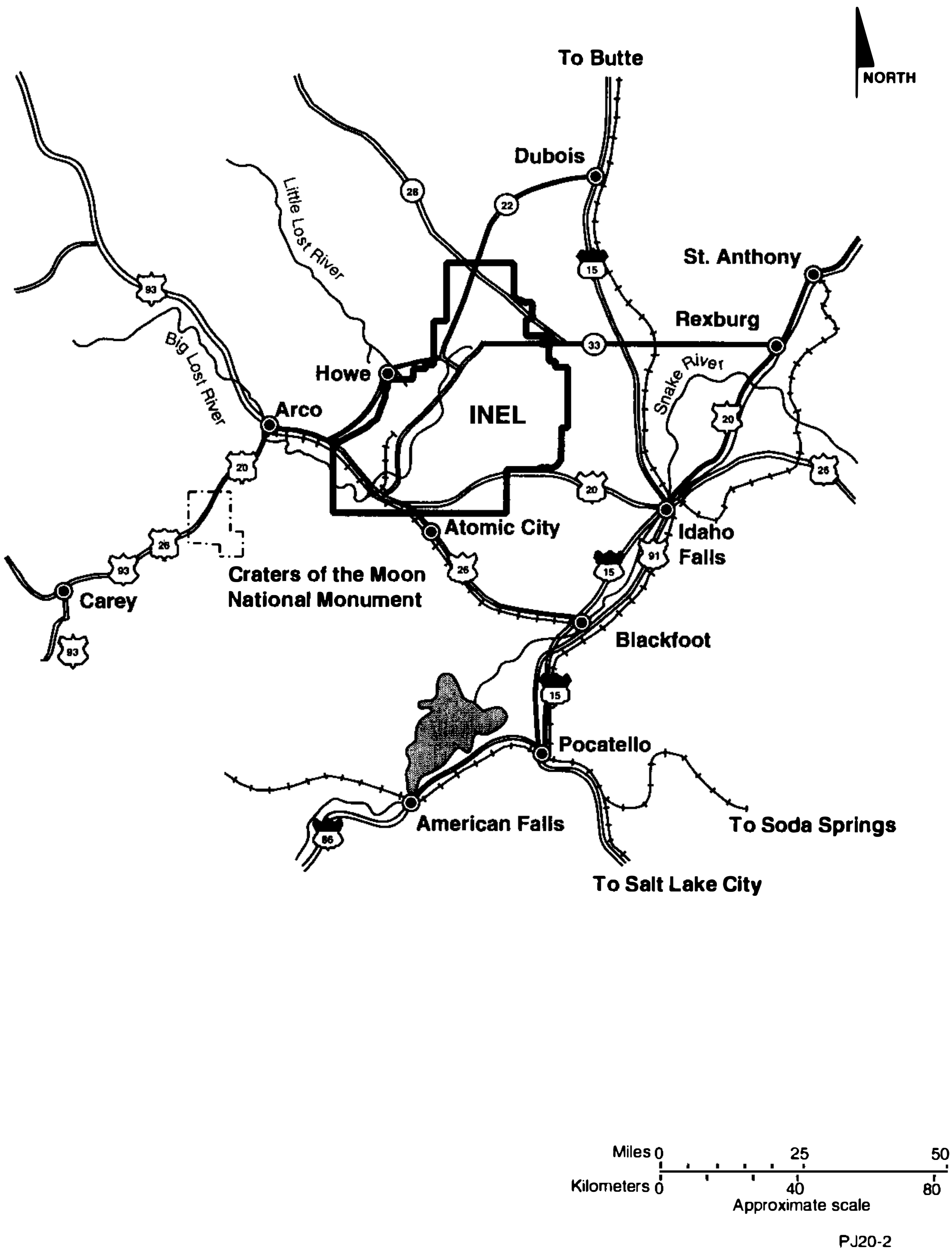


Figure 4.11-1. Transportation routes in the vicinity of the INEL.

Table 4.11-1. Baseline traffic for selected highway segments.[a]

Route	Average daily traffic	Peak hourly traffic[b]
U.S. Highway 20-Idaho Falls to INEL	2,290	344
U.S. Highway 20/26-INEL to Arco	1,500	225
U.S. Highway 26-Blackfoot to INEL	1,190	179
State Route 33 west from Mud Lake	530	80
Interstate 15-Blackfoot to Idaho Falls	9,180	1,380

a. Source: Lehto (1993).
b. Estimated as 15 percent of average daily traffic.

Electronic Technology Center, and DOE Office Buildings. This intersection is congested during peak weekday hours, but it is designed for the current traffic.

4.11.1.3 Transit Modes. Four major modes of transit use the regional highways, community streets, and INEL site roads to transport people and commodities: DOE buses and shuttle vans, DOE motor pool vehicles, commercial trucks, and personal vehicles. Table 4.11-2 summarizes the baseline miles for INEL-related traffic.

Table 4.11-2. Baseline annual vehicle miles traveled for Idaho National Engineering Laboratory-related traffic.[a]

Mode of travel and transportation	Vehicle miles traveled[b]
DOE buses	6,068,200
Other DOE vehicles	9,183,100
Commercial trucks	56,000
Personal vehicles on highways to INEL	7,500,000
TOTAL	22,807,300

a. Source: Lehto (1993).
b. To convert from miles to kilometers, multiply by 1.61.

4.11.2 Railroads

Figure 4.11-1 shows the Union Pacific Railroad lines in southeastern Idaho. Idaho Falls receives railroad freight service from Butte, Montana, to the north, and from Pocatello and Salt Lake City to the south. The Union Pacific Railroad's Blackfoot-to-Arco branch, which crosses the southern portion of the INEL, provides rail service to the site for the shipment of spent nuclear fuel and other waste, bulk commodities, and radioactive materials. This branch connects with a DOE-owned spur line at Scoville Siding, then links with developed INEL areas. Table 4.11-3 lists rail shipments for Fiscal Years 1988 through 1992.

Table 4.11-3. Loaded rail shipments to and from the Idaho National Engineering Laboratory (1988-1992).[a]

Fiscal Year	Inbound	Outbound
1988	63	44
1989	43	19
1990	34	3
1991	18	0
1992	23	0

a. Sources: DOE Shipment Mobility/Accountability Collection System database; Attachment A to Appendix D of Volume 1 of this EIS.

4.11.3 Airports and Air Traffic

Commercial airlines provide Idaho Falls with jet aircraft passenger and cargo service, as well as commuter service to both the Idaho Falls and Pocatello airports. In addition, local charter service is available in Idaho Falls, and private aircraft use the major airport and many other fields in the area. Total landings at the Idaho Falls airport for 1991 and 1992 were 5,367 and 5,598, respectively. The Idaho Falls and Pocatello airports collectively record nearly 7,500 landings annually.

Non-DOE air traffic over the INEL site is limited to altitudes greater than 305 meters (1,000 feet) over buildings and populated areas, and non-DOE aircraft are not permitted to use the site. The primary air traffic at the INEL site is DOE helicopters, which are used for security and emergency purposes. These helicopters have specific operations stations and duties.

4.11.4 Accidents

From 1987 through 1992, the average motor vehicle accident rate was 0.94 accident per million kilometers (1.5 accidents per million miles) for INEL vehicles, which compares with an accident rate of 1.5 accidents per million kilometers (2.4 accidents per million miles) for all DOE complex vehicles and 8 accidents per million kilometers (12.8 accidents per million miles) nationwide for all motor vehicles (Lehto 1993). There are no recorded rail or air accidents associated with the INEL and, to date, no fatal air traffic accidents have involved flights through either the Idaho Falls or Pocatello airports.

4.11.5 Transportation of Waste, Materials, and Spent Nuclear Fuel

Hazardous, radioactive, industrial commercial, and recyclable wastes are transported on the INEL site. Federal and State regulations and requirements govern the transportation of hazardous and radioactive materials (Lehto 1993). Hazardous materials include commercial chemical products and hazardous wastes that are nonradioactive; they are regulated and controlled based on their chemical toxicity. Onsite spent nuclear fuel comes from Argonne National Laboratory - West, the Naval Reactors Facility, and the Advanced Test Reactor; it is transported by truck to various onsite storage and research and development facilities.

This assessment used six years of data (1987 through 1992) to establish a baseline of radiological doses from incident-free, onsite total nonnaval spent nuclear fuel transportation at the INEL. Table 4.11-4 lists the results in terms of cumulative doses (1995-2035) and health effects. These doses do not include onsite naval shipments, which are assessed in Attachment A to Appendix D of Volume 1 of this EIS. The baseline includes no offsite shipments, which are addressed in Appendixes D and I.

Table 4.11-4. Cumulative doses and cancer fatalities from incident-free onsite shipments of nonnaval spent nuclear fuel at the Idaho National Engineering Laboratory for 1995 through 2035.[a,b]

	Estimated collective dose (person-rem)	Estimated cancer fatalities
Occupational	3.4	0.0014
General population	0.087	0.000044

a. Source: Maheras (1993).
b. Onsite naval shipment doses are addressed in Attachment A to Appendix D of Volume 1 of this EIS.

4.12 Occupational and Public Health and Safety

4.12.1 Radiological Health and Safety

DOE Order 5480.11, "Radiation Protection for Occupational Workers" (DOE 1992b), limits the radiation dose that INEL workers can receive to 5 rem per year; administrative controls further limit a worker dose to 2 rem per year, except under unusual circumstances. In addition, DOE has established a comprehensive program, known as ALARA (As Low As Reasonably Achievable), to ensure the reduction of occupational doses to the extent practicable.

The largest fraction of the occupational dose received by INEL workers is from external radiation. Internal radiation doses constitute a small fraction of the occupational dose. Personnel who could receive annual external radiation exposures with measured doses greater than 0.1 rem receive a thermoluminescent dosimeter that they must wear at all times during work on the site. DOE used recorded doses for 1987 to 1991 as a baseline for routine site operations for this EIS. During this period, the INEL monitored about 6,000 workers annually for radiation exposure. About 32 percent of those individuals received measurable radiation doses. Monitoring reports indicate that, from 1987 to 1991, 20 individuals (most of whom were maintenance and construction workers employed by M-K Ferguson at the Idaho Chemical Processing Plant) received annual doses larger than 2 rem (4 individuals in 1987, 1 in 1989, and 15 in 1990).

From 1987 to 1991, the average occupational dose to individuals who had received measurable doses was 0.156 rem per year, resulting in an average collective dose (the number of monitored workers receiving measurable doses was about 32 percent or 1,920) of about 300 person-rem. The resulting number of expected excess latent cancer fatalities would be less than 1 for each year of operation.

This analysis based the doses to the maximally exposed individual and offsite population on baseline radioactive concentrations associated with normal operations. The baseline dose to the maximally exposed individual is 5.6×10^{-2} millirem, which corresponds to a latent fatal cancer probability of 2.8×10^{-8}. The baseline population dose is 7.0×10^{-2} person-rem which, corresponds to a latent fatal cancer incidence of less than 1 (4×10^{-5}) annually and less than 1 (1×10^{-3}) over 40 years.

4.12.2 Nonradiological Exposure and Health Effects

DOE used the air quality data in Table 4.7-2 to evaluate health impacts associated with potential exposure to two compound classes: criteria pollutant and toxic. This analysis has based health effects on air emissions only, and not water pathways, because none of the alternatives would involve the discharge of pollutants to surface waters or the subsurface. Table 4.7-2 lists 5 criteria pollutant and 26 toxic compounds. The classification of two of the toxic compounds (benzene and formaldehyde) as carcinogens was consistent with EPA designations published in the Integrated Risk Information System (IRIS) data base (DOE 1991b). However, this data base does not include sufficient data to perform a quantitative inhalation cancer risk assessment.

To obtain a hazard index, this analysis evaluated toxic and criteria pollutant compound health effects by adding hazard quotients for each compound. The EPA Risk Assessment Guidance for Superfund (EPA 1989) describes this approach. The hazard quotient is the ratio of compound concentration or dose to a Reference Concentration (RfC) or Dose (RfD). For compounds without listed Reference Concentration or Dose values, the analysis used appropriate State of Idaho standards. The use of the noncancer hazard index assumes a level of exposure (standard) below which adverse health effects would be unlikely. The hazard index is not a statistical probability; therefore, it cannot be interpreted as such.

This analysis based toxic and criteria pollutant compound hazard index values for the maximally exposed individual on the maximum concentrations for the compounds at the INEL site boundary, public access roads inside the INEL site boundary, and the Craters of the Moon Wilderness Area. Because the hazard index for criteria pollutants is less than 1, no adverse health effects would be likely from routine operations for either workers or the maximally exposed individual. Because the hazard index for toxic pollutants exceeds 1, the potential for carcinogenic health risks could exist. However, varying spacial and temporal distributions of the concentrations of individual air pollutants make it unlikely that any individual would be exposed to all the pollutants all the time. Since individual hazard indices for the toxic compounds are less than 1, adverse health effects are not expected.

4.12.3 Occupational Health and Safety

Total injury and illness incidence rates at the INEL varied from an annual average of 1.8 to 4.9 per 200,000 work hours from 1987 to 1991. During this time, total lost workday cases ranged from a low of 1 per 200,000 work hours in 1988 and 1989 to a high of 2.6 per 200,000 work hours in

1991. The rates appear higher for 1991 because of a 1990 change in reporting requirements for injuries and illnesses. INEL rates for 1987 to 1989 are below overall DOE rates (2.9 total injury and illness incidence and 1.4 total lost workday cases per 200,000 work hours) and Bureau of Labor Statistics rates (8.5 total injury and illness incidence and 4.0 total lost workday cases per 200,000 work hours). For 1990 and 1991, INEL rates are slightly above overall DOE rates, but below Bureau of Labor Statistics rate.

There were 1,337 total recordable injury and illness cases at the INEL from 1987 to 1991, for an average of 8,385 employees working 79,654,000 hours. Of these cases, 114 (8.5 percent) were occupational illnesses, of which 48 percent were repeated trauma disorders and 30 percent were classified as skin diseases or disorders. One fatality occurred at the INEL between 1987 and 1991 when an employee was struck and killed by a forklift.

4.13 Idaho National Engineering Laboratory Services

This section discusses water, electricity, fuel capacities and consumption, wastewater disposal, and security and emergency protection at INEL facilities.

4.13.1 Water Consumption

A system of about 30 wells, with pumps and storage tanks, provides the water supply for the INEL site. Because of the distance between site facility areas, the water supply system for each facility is independent. The site uses no natural surface water. The City of Idaho Falls water supply system, which includes about 16 wells, provides water to DOE and contractor facilities in the city.

A Water Rights Agreement between DOE and the State of Idaho regulates groundwater use at the INEL site. Under this agreement, INEL has claim to 2,300 liters per second (36,000 gallons per minute) of groundwater, not to exceed 43 billion liters (11 billion gallons) per year (Teel 1993). DOE has not measured the total pumping rate from the aquifer, which would depend on the number of pumps operating. There is a slight possibility that the site could exceed the regulated pumping rate for very short periods, such as during recovery from an extended power outage when many pumps would run to refill depleted storage tanks.

The average INEL site water consumption from 1987 through 1991 was 7.4 billion liters (1.9 billion gallons) per year, based on the cumulative volumes of water withdrawn from the wells (Teel 1993). The projected baseline usage for 1995 will be about 6.5 billion liters (1.7 billion gallons). The estimated average water consumption of Idaho Falls facilities is 300 million liters (80 million gallons) per year.

4.13.2 Electricity Consumption

The Antelope substation supplies commercial electric power to the INEL site through two feeders to the Federally owned Scoville substation. The Scoville substation supplies electric power directly to the INEL electric power distribution system (Teel 1993). The contract with Idaho Power Company to supply electric power to the INEL site provides "up to 45,000 kilowatts monthly" at 13.8 kilovolts (IPC/DOE 1986). Hydroelectric generators along the Snake River in southern Idaho and the Bridger and Valmy coal-fired thermal electric generation plants in southwestern Wyoming and northern

Nevada, respectively, generate the electric power supplied by Idaho Power. The Experimental Breeder Reactor-II can also provide approximately 12 to 15 megavolt-amperes of capacity for the electric power loop (Teel 1993).

The rated capacity of the INEL site power transmission loop line is 124 megavolt-amperes. The peak demand on the system from 1990 through 1993 was about 40 megavolt-amperes, and the average usage was slightly less than 217,000 megawatt-hours per year (Teel 1993). This usage rate should decrease by about 4 percent by 1995.

The INEL facilities in Idaho Falls receive electric power from the City of Idaho Falls, which operates four hydroelectric power generation plants on the Snake River along with substation and distribution facilities. The Bonneville Power Administration, which operates hydroelectric plants on the Columbia River system, supplies supplemental power to the City of Idaho Falls. In 1993, Idaho Falls facilities used 31,500 megawatt-hours of electricity (Teel 1993).

4.13.3 Fuel Consumption

Fuels consumed at the INEL site include several liquid petroleum fuels, coal, and propane. All fuels are transported to the site for storage and use. Natural gas is the only reported fuel consumed at the INEL Idaho Falls facilities; the Intermountain Gas Company provides this fuel through a system of underground lines (Teel 1993).

The average annual fuel consumption at the INEL site from 1990 through 1993 was as follows: fuel oil, 10,578,000 liters (2,795,000 gallons); diesel fuel, 5,690,000 liters (1,500,000 gallons); and propane gas, 568,000 liters (150,000 gallons). The INEL also uses about 8,200 metric tons (9,000 tons) of coal. Fuel storage is provided at each facility and inventories are restocked as necessary. No fossil fuel shortage has ever occurred at the INEL site (Teel 1993).

4.13.4 Wastewater Disposal

Sanitary wastewater systems at the smaller onsite facility areas consist primarily of septic tanks and drain fields. The larger areas, such as Central Facilities Area, Idaho Chemical Processing Plant, and Test Reactor Area, have wastewater treatment facilities. The City of Idaho Falls wastewater treatment system serves the Idaho Falls facilities (Teel 1993).

The average annual wastewater discharge volume at the INEL site from 1989 through 1991 was 537 million liters (142 million gallons). The wastewater from DOE and contractor-operated facilities in Idaho Falls is not metered but is estimated to be 300 million liters (80 million gallons) per year. The primary causes of the difference between water pumped and estimated wastewater discharge are evaporation from ponds and cooling towers, irrigation of landscaped areas, and discharge of unmetered wastewater (Teel 1993). Some industrial wastewater, such as steam condensate, is also discharged to evaporation ponds and injection wells.

4.13.5 Security and Emergency Protection

This section describes the fire protection and prevention, security, and emergency preparedness resources for the INEL site and the surrounding areas. This discussion includes the INEL Fire Department, DOE and INEL Emergency Preparedness, and DOE and INEL Security. DOE established an Emergency Management System that incorporates all applicable requirements for emergency planning, preparedness, and response at the INEL. Each INEL facility must prepare an Emergency Plan that contains detailed contingency plans and emergency procedures.

4.13.5.1 DOE Fire Department. The contractor-operated Fire Department staffs and operates three fire stations on the INEL that support the entire site. Each station has the equipment and expertise to respond to explosions, fires, spills, and medical emergencies. These stations are on the north end at Test Area North, at Argonne National Laboratory-West, and at the Central Facilities Area. Each station has a minimum of one engine company capable of supporting any fire emergency in its assigned area. The Fire Department has a staff of 44 firefighters and 11 support personnel and operates with a minimum critical staff of 7 firefighters at any time. In addition to providing firefighting services, the Fire Department provides the INEL ambulance, emergency medical technician (EMT), and hazardous material response services. The Fire Department has mutual aid agreements with other firefighting organizations, such as the Bureau of Land Management and the Cities of Idaho Falls, Blackfoot, and Arco. Through these agreements, the Idaho Falls Fire Department serves DOE facilities in the City of Idaho Falls.

4.13.5.2 DOE and INEL Emergency Preparedness. Each DOE INEL contractor administers and staffs its own emergency preparedness program under the direction and supervision of DOE. All contractor programs for emergency control and response are compatible. The Warning Communication Center is in the DOE Headquarters building and staffed by the INEL prime contractor with DOE oversight; it is the communication and overall control center for support to onscene commanders in charge of an emergency response. The DOE emergency preparedness system includes mutual aid agreements with all regional county and major city fire departments, police, and medical facilities. Through the agreements, the Idaho Falls emergency preparedness organizations serve DOE facilities in the City of Idaho Falls.

4.13.5.3 DOE and INEL Security. DOE has oversight responsibility for safeguards and security at the INEL. The security program has three categories: security operations, personnel security, and safeguards. The security operations division provides asset protection (classified matter, special nuclear material, facilities, and personnel) and technical security (computer and information). Under this category, DOE administers the INEL protective force, which is supplied by contract. The personnel security staff processes personnel security clearances. The safeguards department is responsible for the management and accountability of special nuclear materials. The INEL protective force, consisting of 200 armed guards and 350 support personnel, provides the onsite personnel who administer the programs. Each INEL contractor has a safeguards and security staff, divided in a similar manner, to manage the security associated with its facilities. Contractor safeguards and security staffs range from about 5 to 60 persons, depending on the size and complexity of the associated facilities. Each staff works with the INEL protective forces.

4.14 Materials and Waste Management

This section summarizes the management of materials and wastes (high-level, transuranic, mixed low-level, low-level, hazardous, industrial and commercial solid wastes and hazardous materials) at the INEL and Idaho Falls facilities, and presents an overview of the current status of the various waste types generated, stored, and disposed at the INEL.

The total amount of waste generated and disposed has been reduced through waste minimization and treatment. The INEL attains waste minimization by reducing or eliminating waste generation, by recycling, and by reducing the volume, toxicity, or mobility of waste before storage or disposal. In addition, the site has achieved volume reduction of radioactive wastes through more intensive surveying, waste segregation, and use of administrative and engineering controls.

The quantitative data presented in this section are from Volume 2 of this EIS, unless otherwise noted.

4.14.1 High-Level Waste

At present, about 11,900 cubic meters (4,970 cubic yards calcine solid and 2,140,000 gallons liquid) of high-level waste are in storage at the INEL Idaho Chemical Processing Plant (see Figure 2-1 for locations of major waste management facilities). This facility blends liquid waste, consisting of aluminum and zirconium wastes from past spent nuclear fuel reprocessing, and sodium-bearing wastes, and processes them through calcination to produce a granular calcine solid. Because of the termination of reprocessing, the site no longer generates liquid high-level waste, with the exception of high-level waste residues. Liquid high-level wastes generated by prior reprocessing activities are solidified at the site. At present, the site generates liquid waste that is not directly the result of reprocessing. The site manages this liquid as high-level waste. The site will calcine the liquid high-level waste that does not contain sodium, and as much sodium-bearing high-level waste as practicable by January 1, 1998, in accordance with the *Amended Order Modifying Order of June 28, 1993*, United States District Court for the District of Idaho, December 22, 1993. The projected 1995 baseline for high-level waste generation is 750 cubic meters (980 cubic yards) annually (EG&G 1993).

4.14.2 Transuranic Waste

About 65,000 cubic meters (85,000 cubic yards) of transuranic and alpha-contaminated low-level wastes are retrievably stored and 62,000 cubic meters (81,000 cubic yards) of transuranic waste (Morton and Hendrickson 1995) have been buried at the Radioactive Waste Management Complex at the INEL. At present, no facilities can dispose of transuranic waste; however, DOE ultimately intends to retrieve, repackage, certify, and ship stored transuranic wastes at the INEL to a potential Federal repository for final disposition. DOE has not determined the disposition of alpha-contaminated low-level waste and buried waste. Since the October 1988 ban by the State of Idaho prohibiting shipments of transuranic waste to the INEL, DOE has shipped only minor amounts of transuranic waste generated on the site to the INEL Radioactive Waste Management Complex for interim storage. At present, there are no treatment facilities for transuranic wastes at the INEL. The projected 1995 baseline for transuranic waste generation is 6 cubic meters (8 cubic yards) annually (EG&G 1993).

4.14.3 Mixed Low-Level Waste

At present, DOE accepts only mixed low-level waste generated at the INEL for treatment and disposal at the INEL. DOE stores mixed low-level waste generated at the INEL at interim storage facilities until treatment systems become available or operational. A total of 1,800 cubic meters (2,400 cubic yards) of mixed low-level waste interim storage capacity is available at the INEL. Current mixed low-level waste interim storage is approximately 1,100 cubic meters (1,400 cubic yards). Treatment technologies exist for much of the mixed low-level waste generated at the INEL, and waste minimization eliminates potential sources of mixed low-level waste before generation. The projected 1995 baseline for mixed low-level waste is 525 cubic meters (687 cubic yards) annually (EG&G 1993).

4.14.4 Low-Level Waste

Through 1991, DOE disposed of 145,000 cubic meters (190,000 cubic yards) of low-level waste at the Radioactive Waste Management Complex. In 1991, the total available low-level waste disposal capacity at the complex was 37,000 cubic meters (48,000 cubic yards). DOE has curtailed low-level waste treatment since 1991 while waiting for updated safety documentation and an environmental impact assessment for the Waste Experimental Reduction Facility. The INEL stores low-level waste awaiting treatment on asphalt or concrete pads at the Waste Experimental Reduction Facility and in radioactive waste storage containers at the generating facilities. The projected 1995 baseline for low-level waste generation is 4,270 cubic meters (5,585 cubic yards) annually (EG&G 1993).

4.14.5 Hazardous Waste

DOE collects hazardous waste generated at the INEL and stores it temporarily at the Hazardous Waste Storage Facility before shipping it off the site. The Hazardous Waste Storage Facility has adequate storage capacity [approximately 64 cubic meters (84 cubic yards)] to manage the quantities of hazardous waste generated at the INEL. The site recycles, reuses, or reprocesses such waste if possible, and might replace some hazardous substances with nonhazardous substances.

4.14.6 Industrial/Commercial Solid Waste

DOE disposes of the industrial and commercial solid waste generated at the site in the INEL Landfill Complex at the Central Facilities Area. The Landfill Complex has approximately 910,000 square meters (225 acres) of land available for solid waste disposal, including the remaining area at Landfill III, which is currently in use. The estimated capacity of the INEL Landfill Complex will be sufficient to dispose of INEL waste for 30 to 50 years; however, capacity of the current excavations will be filled by 1998. DOE has proposed expanding the excavation. Volume 2 of this EIS describes the landfill expansion project. The industrial and commercial solid waste landfill currently in use is in a 48,000-square-meter (12-acre) gravel pit area north of Disposal Area II. DOE does not expect to store solid waste intended for disposal. Waste segregation occurs at each INEL facility so recyclable materials do not enter the solid waste stream. The average annual volume of waste disposed at the Central Facilities Area landfill from 1988 through 1992 was approximately 52,000 cubic meters (68,000 cubic yards) (also the projected 1995 baseline) (EG&G 1993).

4.14.7 Hazardous Materials

The INEL 1993 chemical inventory lists 774 hazardous chemicals. The number and the total weight of hazardous chemicals used on the site and at individual facilities change daily in response to use. The annual Superfund Amendments and Reauthorization Act reports for the INEL facilities include year-to-year inventories.

5. ENVIRONMENTAL CONSEQUENCES

5.1 Overview

This chapter discusses the potential environmental consequences for each spent nuclear fuel management alternative described in Chapter 3. The U.S. Department of Energy (DOE) used the environmental consequence analyses of nonnaval spent nuclear fuel management from Volume 2 as input for this chapter; however, DOE made necessary adjustments to accommodate the differences between Volume 1 and Volume 2 alternatives. In addition, DOE adjusted the 10-year planning horizon for Volume 2 alternatives to 40 years for Volume 1.

As described in Chapter 1, this chapter analyzes only nonnaval DOE actions; however, Section 5.16, "Cumulative Impacts and Impacts from Connected or Similar Actions," includes impacts from the Naval Nuclear Propulsion Program and nonnaval DOE impacts that are cumulative. The Appendix B restriction of analysis to nonnaval actions results in Alternative 2 (options 2a, 2b, and 2c) becoming a single alternative.

Chapter 5 addresses potential impacts from construction and normal operations for each element of the affected environment described in Chapter 4. In addition, it provides potential consequences from accidents and several types of summary information. In cases where the consequence analysis does not result in a distinction among the alternatives, this chapter describes the consequences without division by alternative to avoid needless repetition. Tables 3-4 through 3-6 in Section 3.2 summarize and compare the potential impacts associated with each alternative.

5.2 Land Use

Alternatives 1, 2, 4b(2), and 5a [No Action, Decentralization, Regionalization by Geography (Elsewhere), and Centralization at other DOE sites] would have the least impact on land use, affecting 0.8 acre (0.003 square kilometer); Alternatives 4b(1) [Regionalization by Geography (INEL)] and 5b (Centralization at the INEL) would result in the greatest changes, impacting nearly 31 acres (0.12 square kilometer).

Overall environmental impacts on land use by any of the alternatives would be small because DOE would build new facilities in developed areas that it has already dedicated to industrial use and that previous activities have disturbed. Under all the alternatives, proposed activities would be consistent with the existing land use plans discussed in Section 4.2 and would be similar to uses in existing developed areas on the site. None of the proposed activities would involve land outside the INEL boundaries, and no effects on surrounding land uses or local land use plans should occur.

No onsite land use restrictions due to Native American treaty rights would exist for any of the alternatives described in this EIS. Potential impacts on Native American and other cultural resources are discussed in Section 5.4 (Cultural Resources) and in Appendix L (Environmental Justice).

5.3 Socioeconomics

This section describes the potential effects of the spent nuclear fuel alternatives on the socioeconomic resources of the region of influence described in Section 4.3. Tables 5.3-1 and 5.3-2 list proposed changes in the INEL-related workforce and population. Figure 5.3-1 shows these proposed changes.

5.3.1 Methodology

This section addresses socioeconomic impacts in terms of both direct and secondary employment and population effects. Direct effects are changes in INEL employment that DOE expects to occur under each alternative and include construction and operations phase impacts. Secondary effects include indirect and induced impacts. Indirect effects are impacts to regional businesses and employment resulting from changes in DOE regional purchases or nonpayroll expenditures. Induced effects are impacts to regional businesses and employment that result from changes in payroll spending by affected INEL employees. The total economic impact to the region is the sum of direct and secondary effects.

The bases for the estimated direct impacts in this section are project summary data that DOE developed in cooperation with INEL contractors. Employment impacts represent actual changes in INEL staffing; they do not include changes in staffing due to a reassignment of the existing INEL workforce. The projected decline in baseline INEL activity is not part of any alternative and therefore, a comprehensive analysis of potential impacts was not included. Projected declines in baseline site employment are presented in Figure 5.3-1 in order to provide the reader with a framework for evaluating potential employment and population impacts. This assessment used RIMS II to estimate total employment impacts with multipliers that the U.S. Bureau of Economic Analysis developed specifically for the INEL region of influence. A comprehensive discussion of the methodology is provided in Appendix F-1 of Volume 2. Cumulative impacts on socioeconomic resources in the region are discussed in Section 5.16.

Table 5.3-1. Estimated changes in employment and population for Alternatives 3, 4a, 4b(1) and 5b, 1995 - 2004.[a]

Factor	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Direct employment	0	0	0	0	250	250	375	375	375	375
Secondary employment	0	0	0	0	352	352	528	528	528	528
Total employment change	0	0	0	0	602	602	903	903	903	903
Change in ROI[b] labor force (%)	0.0	0.0	0.0	0.0	0.5	0.5	0.8	0.8	0.8	0.7
Change in ROI employment (%)	0.0	0.0	0.0	0.0	0.6	0.6	0.8	0.8	0.8	0.8
Population change	0	0	0	0	2,027	2,027	3,040	3,040	3,040	3,040
Change in ROI population (%)	0.0	0.0	0.0	0.0	0.8	0.8	1.1	1.1	1.1	1.1

a. Sources: Johnson (1995); USBEA (1993); USBC (1992).
b. ROI = region of influence.

Table 5.3-2. Estimated changes in employment and population for Alternatives 4b(2) and 5a, 1995 - 2004.

Factor	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Direct employment	50	50	0	0	0	0	0	0	0	0
Secondary employment	70	70	0	0	0	0	0	0	0	0
Total employment change	120	120	0	0	0	0	0	0	0	0
Change in ROI[a] labor force (%)	0.1	0.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Change in ROI employment (%)	0.1	0.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Population change	405	405	0	0	0	0	0	0	0	0
Change in ROI population (%)	0.2	0.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

a. Sources: Johnson (1995); USBEA (1993); USBC (1992).
b. ROI = region of influence.

5.3.2 Alternatives 1 and 2 - No Action and Decentralization

Activities associated with Alternatives 1 and 2 would not result in any additional construction or operations jobs at the INEL; therefore, implementation of either of these alternatives would have no impact on socioeconomic resources in the region of influence.

5.3.3 Alternatives 3, 4a, 4b(1), and 5b - 1992/1993 Planning Basis, Regionalization by Fuel Type, Regionalization by Geography (INEL), and Centralization at the INEL

5.3.3.1 Construction. As listed in Table 5.3-1, construction employment under these alternatives would peak during the period from 2001 to 2004 with approximately 375 additional direct jobs per year. When added to the estimated 528 indirect jobs, the total employment impact in the region would be an addition of approximately 903 jobs. Employment would decline to zero by 2008.

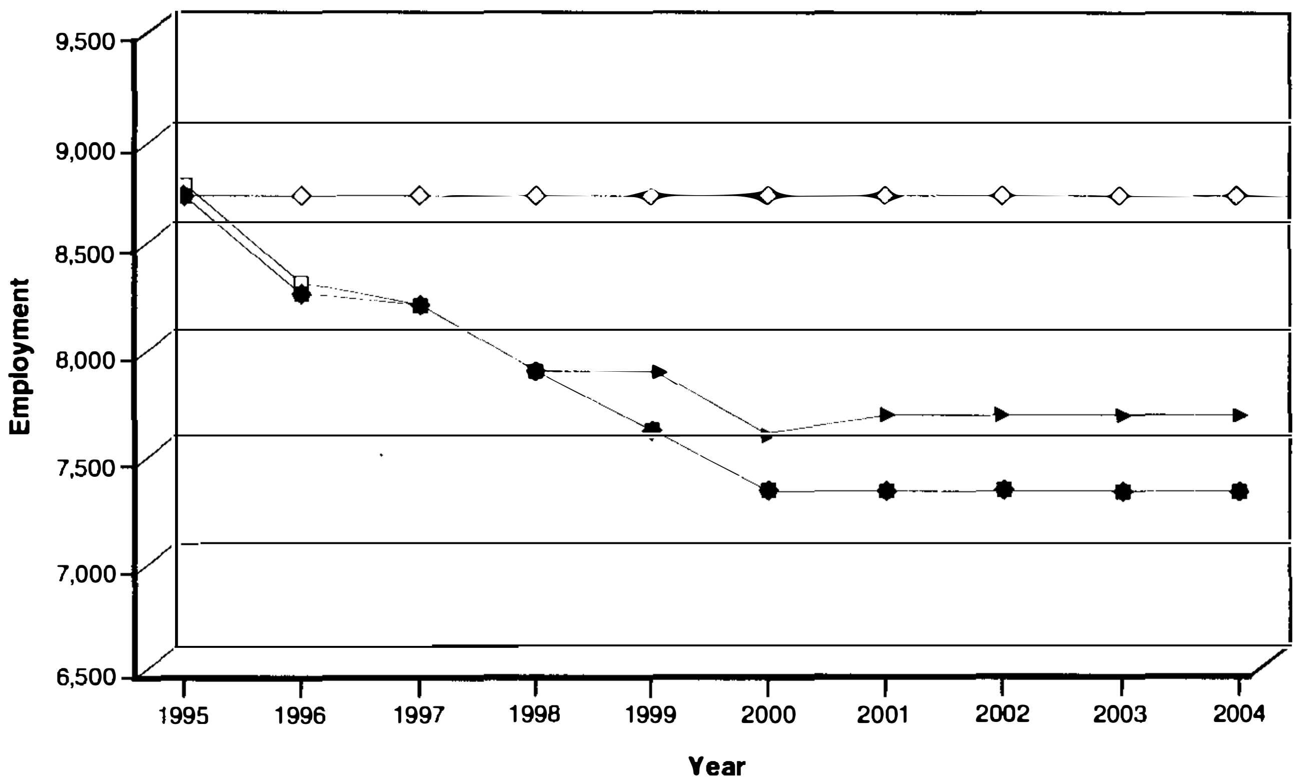
Based on historic data, approximately 97 percent of the new employees who would fill these jobs would live in the seven-county region of influence. As listed in Table 5.3-1, if all new jobs (903) were filled by in-migrants to the region, there would be a 0.8-percent increase in the regional labor force and in regional employment during the peak years. These changes would be minimal and would have no adverse impacts on socioeconomic resources in the region. In fact, although the implementation of any of these alternatives would result in an increase over projected employment levels, as shown in Figure 5.3-1, there would be an overall decline in employment from projected 1995 levels.

Assuming each new employee represented one household and 3.47 persons per household, there would be a corresponding increase in regional population levels of 1.1 percent (approximately 3,000 people). Given this minor change in population, DOE expects potential impacts on the demand for community resources and services such as housing, schools, police, health care, and fire protection to be negligible.

5.3.3.2 Operations. Activities associated with Alternatives 3, 4a, 4b(1), and 5b would not require any additional operations jobs at the INEL. Therefore, the implementation of either of these alternatives would have no impact on socioeconomic resources in the region of influence.

5.3.4 Alternatives 4b(2) and 5a - Regionalization by Geography (Elsewhere) and Centralization at Other DOE Sites

5.3.4.1 Construction. As listed in Table 5.3-2, construction employment under these alternatives would peak during the period from 1995 to 1996 with approximately 50 additional direct jobs per year. When added to the estimated 70 indirect jobs, the total employment impact in the region would be approximately 120 jobs. Employment after 1996 would drop to zero.



Legend:

- ■ Projected site employment as of January 9, 1995.
- □ Alternatives 4b(2) and 5a
- ◆ Alternatives 1 and 2[a]
- ◇ 1995 employment level
- ▶ Alternatives 3, 4b(1), and 5b

a. Alternatives 1 and 2 are the same as the projected site employment.

PJ20-4

Figure 5.3-1. INEL employment by SNF alternative relative to site employment projections.

Based on historic data, approximately 97 percent of the new employees who would fill these jobs would live in the seven-county region of influence. As listed in Table 5.3-2, if all new jobs (120) were filled by in-migrants to the region, there would be a 0.1-percent increase in the regional labor force and in regional employment levels during the peak years. These changes would be minimal and would have no adverse impacts on socioeconomic resources in the region. In fact, although the implementation of any of these alternatives would be an increase over projected employment levels from 1995 to 1996, as shown in Figure 5.3-1, there would be an overall decline in employment from projected 1995 levels.

Assuming each new employee represented one household and 3.47 persons per household, there would be a corresponding increase in regional population levels of 0.2 percent (approximately 400 people). Given this minor change in population, DOE expects potential impacts on the demand for community resources and services such as housing, schools, police, health care, and fire protection to be negligible.

5.3.4.2 Operations. Activities associated with Alternatives 4b(2) and 5a would not result in any additional operations jobs at the INEL. Therefore, the implementation of either of these alternatives would have no impact on socioeconomic resources in the region of influence.

5.4 Cultural Resources

This section summarizes the potential impacts of spent nuclear fuel management activities on cultural resources at the INEL site.

This assessment evaluated both direct and indirect impacts due to the proposed alternatives. At the INEL, direct impacts to archaeological resources usually would be those associated with ground disturbance from construction activities. Direct impacts to existing historic structures could result from demolition, modification, deterioration, isolation from or alteration of the character of the property's setting; or introduction of visual, audible, or atmospheric elements out of character or that alter the property's setting. In addition, indirect impacts to archaeological resources could occur due to an overall increase in activity at the INEL, which could bring a larger workforce closer to significant sites. Direct impacts to traditional resources could occur through land disturbance, vandalism, or changes to the environmental settings of traditional use and sacred areas. Impacts could result from pollution, noise, and contamination that could affect the traditional hunting and gathering areas or the visual or audible settings of sacred areas.

The potential for adverse impacts on cultural resources would be the least under Alternatives 1, 2, 4b(2), and 5a, which would disturb approximately 0.8 acres (0.003 square kilometer). Impacts would be minor because surveys of the area to be disturbed found no eligible cultural resources (Reed et al. 1986; DOE 1993a).

The potential for adverse impacts on cultural resources would be similar under Alternatives 3, 4a, 4b(1), and 5b with the greatest potential under Alternatives 4b(1) and 5b [Regionalization by Geography (INEL) and Centralization at the INEL], which would involve the disturbance of nearly 31 acres (0.12 square kilometer). Again, impacts would be minimal because surveys of the previously disturbed area found no eligible cultural resources (Reed et al. 1986). Under these alternatives, proposed modifications at the Idaho Chemical Processing Plant facilities could adversely affect historically significant structures and could require consultation with the Idaho State Historic Preservation Office (Braun et al. 1993).

The Shoshone-Bannock Tribes are also concerned with the potential impact to important Native American resources from changes in the visual setting, noise, air quality, or water quality. Because activities associated with spent nuclear fuel management would take place within existing facility areas currently engaged in similar activities, DOE does not expect any impacts to important Native

American resources from alteration of the visual setting or noise associated with implementation of any of the alternatives. There could be temporary, minor impacts on air quality from fugitive dust associated with construction activities. Emissions of radionuclides to the air under normal operations would be minor and would be well below applicable standards and guidelines. Under normal operating conditions, radioactive discharges to the soil or directly to the aquifer would not occur.

DOE would minimize the potential for direct and indirect adverse impacts on traditional use resources from pollution, noise, and contamination through compliance with applicable local, state, and Federal laws and regulations. Impact avoidance and other mitigation measures for cultural resources are described in Section 5.20.2.

5.5 Aesthetic and Scenic Resources

None of the alternatives for spent nuclear fuel management at the INEL would have adverse consequences on scenic resources or aesthetics because DOE would confine the proposed projects to developed areas. Although the construction of the proposed facilities would produce fugitive dust that could temporarily affect visibility, the INEL would follow standard construction practices to minimize both erosion and dust generation. Facility operations under each alternative would not produce emissions to the atmosphere that would impact visibility.

5.6 Geology

This section discusses the potential effects of the spent nuclear fuel management alternatives on geologic resources at the INEL site.

Proposed INEL spent nuclear fuel management activities would only have minor localized impacts on the geology of the site for all the alternatives. Direct impacts to geologic resources at the site would be associated with the disturbance or extraction of surface deposits to construct new facilities. These impacts could include excavations into the soil and rock of the site, soil mounding and banking, and the extraction of aggregate materials from gravel and borrow pits on the site. Table 5.6-1 lists estimated extractions of aggregate from site gravel pits for all INEL spent nuclear fuel, environmental restoration, and waste management projects. These values serve to bound the spent nuclear fuel project usage.

A secondary impact to geological resources from construction activities would be the potential for increased soil erosion. DOE would minimize any potential soil erosion by the use of Best Management Practices designed to control stormwater runoff and slope stability.

Table 5.6-1. Estimated INEL gravel/borrow use (cubic meters).[a,b]

Alternative		Estimated Gravel/Borrow Use
1.	No Action	158,000
2.	Decentralization	158,000
3.	1992/1993 Planning Basis	392,000
4a.	Regionalization by Fuel Type	392,000
4b(1)	Regionalization by Geography (INEL)	1,772,000
4b(2)	Regionalization by Geography (Elsewhere)	296,000
5a.	Centralization at other DOE Sites	296,000
5b.	Centralization at the INEL	1,772,000

a. Source: EG&G (1994).
b. To convert cubic meters to cubic yards, multiply by 1.31.

5.7 Air Quality and Related Consequences

This section describes the potential nonradiological and radiological impacts to air quality associated with each alternative. The term "baseline concentrations" is defined as the sum of the concentrations resulting from potential emissions from current operations and those resulting from planned upgrades or modifications that DOE would construct or operate prior to any of the proposed actions described in this EIS. Additional information is provided in Section 5.7 and Appendix F-3 of Volume 2.

5.7.1 Alternative 1 - No Action

5.7.1.1 Nonradiological Air Quality. Construction activities associated with this alternative would be limited to upgrading an existing facility. Potential impacts to air quality from construction activities would include fugitive dust and exhaust emissions from support equipment. DOE assessed the impacts from construction using the EPA Fugitive Dust Model (FDM) (Winges 1992). The modeling results showed that the expected construction-related air quality impacts should be temporary and highly localized.

Minimal spent nuclear fuel activities would occur under this alternative. Therefore, DOE expects that the ambient concentrations levels from normal operations would be similar to those from baseline. Table 4.7-1 lists nonradioactive emissions from normal operations. Tables 5.7-1 and 5.7-2 list the maximum potential concentrations for the proposed alternatives; they are all below applicable standards and guidelines. Ambient concentrations from Alternative 1 activities will be below applicable standards and guidelines.

5.7.1.2 Radiological Air Quality. No radiological impacts to the environment would result from construction activities.

No additional facilities that would be in operation for this alternative would produce radionuclide emissions. Therefore, for normal operations, doses to the maximally exposed individual, the population, and workers would be equivalent to baseline doses, as listed in Table 5.7-3. Table 5.7-4 lists associated emission rates.

Table 5.7-1. Maximum impacts to nonradiological air quality from spent nuclear fuel - criteria pollutants.[a,b]

Pollutant	Averaging time	Applicable standard ($\mu g/m^3$)	Maximum baseline concentration ($\mu g/m^3$)	Baseline plus maximum alternative[c] ($\mu g/m^3$)	Percent of standard
Carbon monoxide	1-hr	40,000	610	610	1.5
	8-hr	10,000	280	280	2.8
Nitrogen dioxide	Annual	100	4	4	4
Lead	Quarterly	1.5	0.001	0.001	<0.1
Particulate matter (PM_{10})	24-hr	150	80	80	53
	Annual	50	5	5	10
Sulfur dioxide	3-hr	1,300	580	580	45
	24-hr	365	140	140	38
	Annual	80	6	6	7.5

a. Source: Section 5.7 of Volume 2 of this EIS and Belanger et al. (1995).
b. Listed concentrations are the maximum of those calculated at the INEL site boundary, public access roads inside the INEL site boundary, and the Craters of the Moon Wilderness Area.
c. The listed concentrations are the maximums for any of the proposed alternatives.

Table 5.7-2. Maximum impacts to nonradiological air quality from spent nuclear fuel - toxic air pollutants.[a,b]

Pollutant	Averaging time	Applicable standard ($\mu g/m^3$)	Maximum baseline concentration ($\mu g/m^3$)	Impact from maximum alternative[c] ($\mu g/m^3$)	Percent of standard[d]
Ammonia	Annual	1.8×10^{2}	6.0×10^{0}	1.8×10^{0}	1
Benzene	Annual	1.2×10^{-1}	2.9×10^{-2}	2.3×10^{-2}	19
Formaldehyde	Annual	7.7×10^{-2}	1.2×10^{-2}	4.4×10^{-2}	57
Methyl isobutyl ketone	Annual	2.1×10^{3}	(e)	2.6×10^{1}	1
Hydrofluoric acid	Annual	2.5×10^{1}	(e)	1.8×10^{-2}	<0.1
Tributylphosphate	Annual	2.5×10^{1}	(e)	6.1×10^{-2}	0.2

a. Source: Section 5.7 of Volume 2 of this EIS and Raudsep (1995).
b. Listed concentrations are the maximum of those calculated at the INEL site boundary, public access roads inside the INEL site boundary, and the Craters of the Moon Wilderness Area.
c. The listed concentrations are the maximums for any of the proposed alternatives, plus new or modified sources expected to become operational after May 1, 1994.
d. In accordance with State of Idaho regulations for toxic air pollutants, the percent of standard is calculated based on concentrations resulting from the alternatives and from new or modified sources that have become operational since May 1, 1994.
e. Baseline concentrations for these pollutants were not analyzed because their emissions were below screening levels.

Table 5.7-3. Annual dose increments by alternative in comparison to the baseline.[a]

Alternative		INEL worker (millirem)	Maximally exposed individual (millirem)	Population (person-rem)[b]
Baseline		4.3×10^{0}[c]	5.6×10^{-2}	3.4×10^{-1}
1.	No Action	3.3×10^{-4}	3.5×10^{-3}	1.0×10^{-1}
2.	Decentralization	3.3×10^{-4}	3.5×10^{-3}	1.0×10^{-1}
3.	1992/1993 Planning Basis[c]	3.3×10^{-3}	8.0×10^{-3}	1.9×10^{-1}
4a.	Regionalization by Fuel Type	3.3×10^{-3}	8.0×10^{-3}	1.9×10^{-1}
4b(1).	Regionalization by Geography (INEL)[d]	4.2×10^{-3}	4.8×10^{-2}	3.9×10^{-1}
4b(2).	Regionalization by Geography (Elsewhere)	7.0×10^{-5}	3.9×10^{-3}	8.3×10^{-2}
5a.	Centralization at Other DOE Sites	7.0×10^{-5}	3.9×10^{-3}	8.3×10^{-2}
5b.	Centralization at the INEL	4.2×10^{-3}	4.8×10^{-2}	3.9×10^{-1}

a. Source: Section 5.7 of Volume 2 of this EIS.
b. Population dose is calculated based on the projected population in 2000 or 2010 whichever is higher.
c. Baseline worker dose includes the maximum projected operation of the portable water treatment unit at the Power Burst Facility area. However, the operation would be temporary (1 to 2 years) and is not representative of a permanent increase in the baseline. If this facility were not included, the baseline dose to the worker would be about 0.2 millirem per year.
d. Alternative 4b(1) doses are slightly less than Alternative 5b doses.

5.7.2 Alternative 2 - Decentralization

5.7.2.1 Nonradiological Air Quality. Potential impacts to air quality from construction activities would include fugitive dust and exhaust emissions from support equipment. The modeling assessment showed that the expected construction-related air quality impacts should be temporary and highly localized.

Emissions resulting from normal operations under this alternative would include baseline emissions and those resulting from the startup of the proposed facilities. Emission rates associated with startup would be less than 1 percent of those from normal operations. Tables 5.7-1 and 5.7-2 list the maximum concentrations predicted for the proposed alternatives. Ambient concentrations from Alternative 2 activities would be below applicable standards and guidelines.

Table 5.7-4. Radionuclide emissions by alternative for spent nuclear fuel projects.[a]

Project and Location	Associated Alternative	Radionuclides and Emission Rates (Ci/yr) H-3/ C-14	Co-60	Kr-85	Xe-131m/ Xe-133	Sr-90/ Y-90	Sb-125	I-129/ I-131	Cs-134 Cs-137	Plutonium	Am-241	Others
TAN Pool Fuel Transfer Project	1, 2, 3, 4a											
a. Drying operations	4b(1), 5b	9.6×10^{2}	-	-	-	2.9×10^{-2}	-	3.4×10^{-2}	-	6.6×10^{-4}	2.2×10^{-4}	-
b. Storage operations		3.9×10^{-1}	-	-	-	-	-	-	-	-	-	-
(Test Area North)												
Additional Increased Rack Capacity (Idaho Chemical Processing Plant)	3, 4a, 4b(1), 5b	2.0×10^{-1}	1.2×10^{-8}	-	-	3.8×10^{-7}	1.0×10^{-4}	-	1.3×10^{-5}	-	-	3.1×10^{-6}
Dry Fuels Storage Facility (Idaho Chemical Processing Plant)	3, 4a, 4b(1), 4b(2), 5a, 5b	1.8×10^{-2}	1.9×10^{-6}	-	-	1.8×10^{-5}	2.2×10^{-3}	4.2×10^{-3}	6.8×10^{-7}	2.6×10^{-7}	-	1.9×10^{-5}
Fort St. Vrain Spent Fuel Storage (Idaho Chemical Processing Plant)	3, 4a, 4b(1), 5b	-	5.6×10^{-8}	-	-	1.8×10^{-6}	-	-	2.4×10^{-7}	5.6×10^{-7}	-	2.4×10^{-7}
Increased Rack Capacity (Idaho Chemical Processing Plant)	3, 4a, 4b(1), 5b	2.0×10^{-1}	1.2×10^{-8}	-	-	3.8×10^{-7}	1.0×10^{-4}	-	1.3×10^{-5}	-	-	3.1×10^{-6}
EBR-II Blanket Treatment (Argonne National Laboratory - West)	3, 4a, 4b(1), 5b	1.6×10^{2}	-	4.9×10^{3}	5.1×10^{1}	-	-	-	-	-	-	-
Electrometallurgical Process Demonstration Project (Argonne National Laboratory - West)	3, 4a, 4b(1), 4b(2), 5a, 5b	8.4×10^{2}	-	1.4×10^{4}	1.3×10^{2}	-	-	-	-	-	-	-
Spent Fuel Processing Facility	4b(1), 5b	3.1×10^{3}	1.9×10^{-6}	5.0×10^{5}	-	5.8×10^{-2}	1.6×10^{1}	4.4×10^{-1}	1.8×10^{-1}	7.7×10^{-3}	-	2.1×10^{-1}

a. Source: Appendix F-3 of Volume 2 of this EIS.

5.7.2.2 Radiological Air Quality. No radiological impacts to the environment would result from construction activities.

Emissions resulting from normal operations under this alternative would include the baseline emissions and those resulting from the startup of the proposed facilities. Table 5.7-4 lists emission rates for the spent nuclear fuel alternatives, including Decentralization. Table 5.7-3 lists the resulting doses to the maximally exposed individual, the population, and workers. These values are small in comparison to the National Emission Standards for Hazardous Air Pollutants dose limit of 10 millirem per year, the dose limit received from background sources of 351 millirem per year, and the population dose from background sources of 40,000 person-rem.

5.7.3 Alternative 3 - 1992/1993 Planning Basis

5.7.3.1 Nonradiological Air Quality. Potential impacts to air quality from construction activities would include fugitive dust and exhaust emissions from support equipment. The modeling assessment showed that expected construction-related air quality impacts should be temporary and highly localized.

Emissions resulting from normal operations under this alternative would include baseline emissions and those resulting from the proposed facilities. Emission rates associated with startup would be less than 1 percent of those from normal operations. Tables 5.7-1 and 5.7-2 list the maximum potential concentrations for the proposed alternatives. Ambient concentrations from Alternative 3 activities would be below applicable standards and guidelines.

5.7.3.2 Radiological Air Quality. No radiological impacts to the environment would result from construction activities.

Emissions resulting from normal operations under this alternative would include baseline emissions and those resulting from the startup of the proposed facilities. Table 5.7-4 lists emission rates for the spent nuclear fuel alternatives. Table 5.7-3 lists the resulting doses to the maximally exposed individual, the population, and workers. These values are small in comparison to the National Emission Standards for Hazardous Air Pollutants dose limit of 10 millirem per year, the dose limit received from background sources of 351 millirem per year, and the population dose from background sources of 40,000 person-rem.

5.7.4 Alternative 4a - Regionalization by Fuel Type

5.7.4.1 Nonradiological Air Quality. Potential impacts to air quality from construction activities would include fugitive dust and exhaust emissions from support equipment. The modeling assessment showed that the expected construction-related air quality impacts should be temporary and highly localized.

Emissions resulting from normal operation under this alternative would include baseline emissions and those resulting from the startup of the proposed facilities. Emission rates associated with startup would be less than 1 percent of those from normal operations. Tables 5.7-1 and 5.7-2 list the maximum potential concentrations for the proposed alternatives. Ambient concentrations from Alternative 4 activities would be below applicable standards and guidelines.

5.7.4.2 Radiological Air Quality. No radiological impacts to the environment would result from construction activities.

Emissions resulting from normal operation under this alternative would include baseline emissions and those resulting from the proposed facilities. Table 5.7-4 lists emission rates for spent nuclear fuel alternatives including Regionalization. Table 5.7-3 lists the resulting doses to the maximally exposed individual, the population, and workers. These values are small in comparison to the National Emission Standards for Hazardous Air Pollutants dose limit of 10 millirem per year, the dose limit received from background sources of 351 millirem per year, and the population dose from background sources of 40,000 person-rem.

5.7.5 Alternative 4b(1) - Regionalization by Geography (INEL)

5.7.5.1 Nonradiological Air Quality. Potential impacts to air quality from construction activities would include fugitive dust and exhaust emissions from support equipment. The modeling assessment showed that the expected construction-related air quality impacts should be temporary and highly localized.

Emissions resulting from normal operation under this alternative would include baseline emissions and those resulting from the startup of the proposed facilities. Emission rates associated with startup would be less than 1 percent of those from normal operations. Tables 5.7-1 and 5.7-2 list the maximum potential concentrations from the proposed alternatives. Ambient concentrations from Alternative 4b(1) activities would be below applicable standards and guidelines.

5.7.5.2 Radiological Air Quality. No radiological impacts to the environment would result from construction activities.

Emissions resulting from normal operation under this alternative would include baseline emissions and those resulting from the proposed facilities. Table 5.7-4 lists associated emission rates for spent nuclear fuel alternatives including Regionalization by Geography (INEL). Table 5.7-3 lists resulting doses to the maximally exposed individual, the population, and workers. These values are small in comparison to the National Emission Standards for Hazardous Air Pollutants dose limit of 10 millirem per year, the dose limit received from background sources of 351 millirem per year, and the population dose from background sources of 40,000 person-rem.

5.7.6 Alternative 4b(2) - Regionalization by Geography (Elsewhere)

5.7.6.1 Nonradiological Air Quality. Potential impacts to air quality from construction activities would include fugitive dust and exhaust emissions from support equipment. The modeling assessment showed that the expected construction-related air quality impacts should be temporary and highly localized.

Emissions resulting from normal operation under this alternative would include baseline emissions and those resulting from the startup of the proposed facilities. Emission rates associated with startup would be less than 1 percent of those from normal operations. Tables 5.7-1 and 5.7-2 list the maximum potential concentrations from the proposed alternatives. Ambient concentrations from Alternative 4b(2) activities would be below applicable standards and guidelines.

5.7.6.2 Radiological Air Quality. No radiological impacts to the environment would result from construction activities.

Emissions resulting from normal operation under this alternative would include baseline emissions and those resulting from the proposed facilities. Table 5.7-4 lists associated emission rates for spent nuclear fuel alternatives including Regionalization by Geography (Elsewhere). Table 5.7-3 lists resulting doses to the maximally exposed individual, the population, and workers. These values are small in comparison to the National Emission Standards for Hazardous Air Pollutants dose limit of

10 millirem per year, the dose limit received from background sources of 351 millirem per year, and the population dose from background sources of 40,000 person-rem.

5.7.7 Alternative 5a - Centralization at Other DOE Sites

5.7.7.1 Nonradiological Air Quality. Potential impacts to air quality from construction activities would include fugitive dust and exhaust emissions from support equipment. The modeling assessment showed that the expected construction-related air quality impacts should be temporary and highly localized.

Emissions resulting from normal operation under this alternative would include baseline emissions and those resulting from the startup of the proposed facilities. Emission rates associated with startup would be less than 1 percent of those from normal operations. Tables 5.7-1 and 5.7-2 list the maximum potential concentrations from the proposed alternatives. Ambient concentrations from Alternative 5a activities would be below applicable standards and guidelines.

5.7.7.2 Radiological Air Quality. No radiological impacts to the environment would result from construction activities.

Emissions resulting from normal operation under this alternative would include baseline emissions and those resulting from the proposed facilities. Table 5.7-4 lists associated emission rates for spent nuclear fuel alternatives including Centralization at other DOE sites. Table 5.7-3 lists resulting doses to the maximally exposed individual, the population, and workers. These values are small in comparison to the National Emission Standards for Hazardous Air Pollutants dose limit of 10 millirem per year, the dose limit received from background sources of 351 millirem per year, and the population dose from background sources of 40,000 person-rem.

5.7.8 Alternative 5b - Centralization at the INEL

5.7.8.1 Nonradiological Air Quality. Potential impacts to air quality from construction activities would include fugitive dust and exhaust emissions from support equipment. The modeling assessment showed that the expected construction-related air quality impacts should be temporary and highly localized.

Emissions resulting from normal operation under this alternative would include baseline emissions and those resulting from the proposed facilities. Emission rates associated with the startup of the proposed facilities would be less than 1 percent of those from normal operations. Tables 5.7-1 and 5.7-2 list the maximum potential concentrations from the proposed alternatives. Ambient concentrations from Alternative 5b activities would be below applicable standards and guidelines.

5.7.8.2 Radiological Air Quality. No radiological impacts to the environment would result from construction activities.

Emissions resulting from normal operation under this alternative would include baseline emissions and those resulting from startup of the proposed facilities. Table 5.7-4 lists associated emission rates for spent nuclear fuel alternatives including Centralization at the INEL. Table 5.7-3 lists resulting doses to the maximally exposed individual, the population, and workers. These values are small in comparison to the National Emission Standards for Hazardous Air Pollutants dose limit of 10 millirem per year, the dose limit received from background sources of 351 millirem per year, and the population dose from background sources of 40,000 person-rem.

5.8 Water Resources and Related Consequences

This section discusses potential environmental consequences to water resources under the five spent nuclear fuel management alternatives. DOE evaluated each alternative with respect to its impacts on water quality (both surface and subsurface water), water use, and human health.

Any liquid effluents from facilities proposed for the spent nuclear fuel alternatives would be in tanks or lined evaporation basins. Under normal operating conditions, radioactive discharges to the soil or directly to the aquifer would not occur. Creed (1994) presents spent nuclear fuel water quality data for the analysis of the potential impacts resulting from a hypothetical leak of 20 liters (5 gallons) per day from secondary containment around the SNF storage pools during operations. Arnett (1994) addresses the effects that this leak could have on the quality of subsurface water resources. Preliminary results indicate that there will be no contaminants above maximum contaminant levels at the INEL boundary resulting from the postulated operational leak. Some storage pools have had leakage in the past. However, based on the bounding accident scenario for high-level waste tank failure, leakage during the implementation of the selected spent nuclear fuel management alternative would cause negligible impacts to water resources (Bowman 1994). None of the proposed alternatives for the management of spent nuclear fuel would result in any renewed discharges to infiltration ponds. Section 5.15 discusses potential releases of hazardous or radioactive liquids as a result of accidents.

With respect to water usage, Alternative 4b(1) [Regionalization by Geography (INEL)] and Alternative 5b (Centralization at the INEL) would consume the largest volume of water--1.5 million cubic meters (400 million gallons) over 40 years. The greatest water consumption rate for these alternatives would be 50,000 cubic meters (13 million gallons) per year (Hendrickson 1995). This incremental usage would represent approximately a 0.7 percent increase over the total average withdrawal rate at the INEL of 7.4 million cubic meters (1.9 billion gallons) per year. The INEL's consumptive use water right is 43 million cubic meters (11.4 billion gallons) per year. Therefore, Alternatives 4b(1) and 5b would have negligible impact on the quantity of water in the Eastern Snake River Plain Aquifer.

5.9 Ecology

DOE expects that construction impacts, which would include the loss of some wildlife habitat due to land clearing and facility development, would be greatest under Alternative 4b(1) [Regionalization by Geography (INEL)] and Alternative 5b (Centralization at the INEL). Because this construction activity would take place either within the boundaries of heavily developed areas or adjacent to those areas, it would have minimal impact on ecological resources. However, construction activities could provide opportunities for the spread of exotic plant species (e.g., cheatgrass and Russian thistle).

There would be no construction impacts to wetlands, which would be excluded from development, and impacts to threatened and endangered species would be unlikely, given the location (previously-developed areas) and the maximum size [approximately 31 acres (0.125 square kilometers)] of the affected area. Construction activities at the INEL probably would not affect either of the endangered species identified in Section 4.9.3 (the bald eagle and peregrine falcon). Both of these birds of prey are associated with riparian areas, wetlands, and larger bodies of water (e.g., reservoirs) and inhabit dry upland areas only temporarily when migrating (National Geographic Society 1987). Disturbance to other sensitive (but not Federally-listed) species identified in Section 4.9.3 (e.g., the burrowing owl, northern goshawk, ferruginous hawk, Swainson's hawk, gyrfalcon, Townsend's western big-eared bat, and pygmy rabbit) would be possible but unlikely, given the scale of the planned construction. Any impacts would be negligible and short lived, lasting only as long as the construction activities.

Representative impacts from operations would include the disturbance and displacement of animals (such as the pronghorn) caused by the movement and noise of personnel, equipment, and vehicles. Such impacts would be greatest under Alternative 4b(1) [Regionalization by Geography (INEL)] and Alternative 5b (Centralization at INEL), which would involve a generally higher level of operational activity; however, these impacts would be minor under all the proposed alternatives.

5.10 Noise

As discussed in Section 4.10, noises generated on the INEL do not travel off the site at levels that affect the general population. Therefore, INEL noise impacts for each alternative would be limited to those resulting from the transportation of personnel and materials to and from the site that would affect nearby communities, and from onsite sources that could affect wildlife near those sources.

Transportation noises would be a function of the size of the workforce (e.g., an increased workforce would result in increased employee traffic and corresponding increases in deliveries by truck and rail; a decreased workforce would result in decreased employee traffic and corresponding decreases in deliveries). This analysis of traffic noise considered railroad noise and noise from major roadways that provide access to the INEL. DOE does not expect the number of freight trains per day in the region and through the site to change as a result of any of the alternatives. Rail shipments of spent nuclear fuel, regardless of the alternative, would be a small fraction of the rail traffic on the Blackfoot-to-Arco Branch of the Union Pacific System line that crosses the INEL. The vehicles that transport employees and personnel on roads would be the principal source of community noise impacts near the INEL.

This analysis used the day-night average sound level to assess community noise, as suggested by the EPA (EPA 1974, 1982) and the Federal Interagency Committee on Noise (FICON 1992). The analysis based its estimate of the change in day-night average sound level from the baseline noise level for each alternative on projected changes in employment and traffic levels. The analysis also considers the combination of construction and operation employment. The baseline noise level is comparable to that for the No-Action alternative. Section 4.10 discusses levels representative of the No-Action alternative. The traffic noise analysis considered U.S. Highway 20, which employees use to access the INEL from Idaho Falls. Changes in noise level below 3 decibels probably would not result in a change in community reaction (FICON 1992).

The new employment associated with each alternative is a small percentage of the total onsite workforce. The maximum new employment of about 375 INEL onsite jobs would occur with Alternatives 3, 4a, 4b(1), and 5b during the peak construction period beginning in 2001 (see Section 5.3, Socioeconomics). No new operations employment is projected for any of the alternatives except Alternatives 4b(1) and 5b for which there would be 25 new jobs beginning in 2007. The cumulative onsite workforce under each alternative would be greatest in 1995 and would decrease thereafter. The peak cumulative onsite workforce for Alternatives 4b(2) and 5a would increase in 1995 by less than 1 percent compared to the No-Action baseline. There would be a corresponding increase in private vehicle and truck trips to the site. The day-night sound level (DNL) at 15 meters (50 feet) from the roads that provide access to the INEL probably would increase by less than 1 decibel. The peak cumulative onsite workforce for Alternative 2 in 1995 would be the same as that for the No-Action baseline.

For any of the alternatives, truck activity would consist of a few trips per day to and from the site carrying spent nuclear fuel. This increase in truck trips would not result in a perceptible increase in traffic noise levels along the routes to the INEL. The day-night average sound level along U.S. Highway 20 and other access routes probably would decrease slightly as a result of the anticipated overall decrease in employment levels at the INEL. DOE expects no change in the community reaction to noise along this route and other access routes. No mitigation efforts would be required.

5.11 Traffic and Transportation

5.11.1 Introduction

Spent nuclear fuel management activities involve the transportation of spent nuclear fuel inside the boundaries of the INEL (onsite) and on highways and rail systems outside the boundaries of the INEL (offsite). This section summarizes the methods of analysis used to determine the environmental consequences of onsite transportation of nonnaval spent nuclear fuel under normal conditions (incident-free) and of transportation accidents. The impacts include doses and health effects. Appendices D and I of Volume 1 address consequences of shipments to or from the INEL that involve other DOE sites and spent nuclear fuel-related locations.

5.11.2 Methodology

5.11.2.1 Incident-Free Transportation. Radiological impacts were determined for two groups of people during normal incident-free transportation: (1) crewmen (drivers) and (2) members of the public. Members of the public are persons sharing the transport link (on-link). On-link doses were determined for onsite shipments because members of the public have access to the majority of the roads on the INEL. Radiological impacts were calculated using the RADTRAN 4 (Neuhauser and Kanipe 1992) and RISKIND (Yuan et al. 1993) computer codes.

The magnitude of the incident-free dose depends mainly on the Transport Index of the shipment and the on-link vehicle densities. The Transport Index is defined as the dose rate at 1 meter (3.28 feet) from the surface of a radioactive package; it is measured in millirem per hour. Spent nuclear fuel was assigned a dose rate of 14 millirem per hour at 1 meter from the shipping container. This dose rate yielded a dose rate of 10 millirem per hour at 2 meters (6.56 feet) from the edge of the transport vehicle, which is the regulatory limit for an exclusive use vehicle (see Madsen et al. 1986).

Radiological doses were converted to cancer fatalities using risk conversion factors of 5.0×10^{-4} fatal cancer per person-rem for members of the public and 4.0×10^{-4} fatal cancers per person-rem for workers. These risk conversion factors are from Publication 60 of the International Commission on Radiological Protection (ICRP 1991).

Because the onsite transportation of spent nuclear fuel at the INEL is considered rural, no incident-free nonradiological risk (from exhaust emissions and dust resuspension) was calculated.

5.11.2.2 Accidents. The doses of the maximum reasonably foreseeable onsite spent nuclear fuel transportation accident were calculated using the RISKIND computer code. Doses were analyzed for generic rural and suburban population densities, assuming 6 persons per square kilometer for rural areas and 719 persons per square kilometer for suburban areas. Areas within 80 kilometers (50 miles) of INEL have population densities between rural and suburban but are closer to the generic rural population density. Doses were also assessed under both neutral and stable atmospheric conditions. Radiation doses calculated were used to estimate the potential for fatal cancers in the exposed population using risk factors developed by the International Commission on Radiological Protection (ICRP 1991).

The probability of the maximum reasonably foreseeable onsite spent nuclear fuel transportation accident was estimated taking into account spent nuclear fuel handling procedures within the Advanced Test Reactor facility as well as factors related to transportation of the spent nuclear fuel. For this accident to occur, errors must occur in loading the wrong spent nuclear fuel into the shipping cask, radiation surveys of the loaded cask fail to detect abnormally high radiation levels, the transport vehicle must breakdown or rollover during the short transit between the Advanced Test Reactor and the Idaho Chemical Processing Plant, and operators fail to ensure that adequate cooling water is maintained inside the cask. The estimated probability of this accident is no greater than once in a million years.

The risk of the onsite spent nuclear fuel transportation accident was estimated by multiplying the accident doses by the accident probability, taking into account the probability of the atmospheric conditions used. The resulting risk value gives a bounding estimate of the annual probability of fatal cancers occurring in the local population due to onsite spent nuclear fuel transportation accidents.

5.11.3 Onsite Spent Nuclear Fuel Shipments

For each spent nuclear fuel management alternative, a small number of onsite DOE spent nuclear fuel shipments would be likely each year as a result of continuing reactor operations at the Advanced Test Reactor and the Experimental Breeder Reactor-II. The alternatives would not affect the operation of these two facilities, thus the shipments between these facilities and the Idaho Chemical Processing Plant, integrated over 40 years, would be the same for each spent nuclear fuel management alternative.

Spent nuclear fuel shipments to the Idaho Chemical Processing Plant from four locations on the INEL (including the Test Reactor Area, Argonne National Laboratory-West, Test Area North, and Power Burst Facility) were evaluated. The number of shipments would not change with alternatives because DOE plans to ship all spent nuclear fuel to the Idaho Chemical Processing Plant. Alternatives that would ship spent nuclear fuel off the site under Regionalization [Alternatives 4a, 4b(1) and 4b(2)] and Centralization (Altemtives 5a and 5b) would ship it first to the Idaho Chemical Processing Plant for canning or other stabilization prior to shipment. DOE estimated the total projected number of shipments over 40 years of operation (1995-2035) from each facility from either historic records or current inventories. DOE based the projected number of shipments for Test Reactor Area and Argonne National Laboratory-West to the Idaho Chemical Processing Plant on historic records for 1987 through 1992, and the doses reflect shipments for 1995 through 2035. The projected number of shipments from Test Area North would include Three Mile Island canisters, Loss of Fluid Test fuel, special case commercial fuel, and non-fuel-bearing components stored in the Test Area North pool. The projected number of shipments from the Power Burst Facility includes all spent nuclear fuel stored at that facility.

Onsite shipments would include those that originated and ended on the INEL site. Shipments that originate or terminate at non-INEL facilities are offsite shipments. Appendixes D and I describe the consequences of naval and DOE offsite spent fuel shipments, respectively. Movements of spent nuclear fuel inside (INEL) facility fences (e.g., from the CPP-603 Underwater Storage Facility to the Fuel Storage Area) are operational transfers, not onsite shipments; therefore, this section does not consider such shipments.

5.11.4 Incident-Free Impacts

The occupational and general population collective doses from onsite spent nuclear fuel shipments and the resulting incidence of latent cancer fatalities were calculated. The results are the same regardless of alternative. Occupational radiation exposure would potentially be 3.4 person-rem, resulting in 0.0014 latent cancer fatalities. General population exposure would potentially be 0.088 person-rem, resulting in 0.000044 latent cancer fatalities.

In addition to collective radiation exposure, the maximally exposed individual doses due to INEL onsite SNF shipments were calculated for a driver (occupational exposure), a person following a single shipment, and a person standing beside the road as a single shipment passes by (general member of the public). The calculated dose to a driver would be 1.7 rem, assuming that person drove all shipments over 40 years. The calculated maximally exposed individual dose to a person following a single shipment covering the longest distance from Test Area North to the Idaho Chemical Processing Plant would be 0.015 millirem, and to a person exposed to passing shipment at a distance of 1 meter (3.28 feet), the dose would be 0.0014 millirem (Maheras 1995).

Traffic impacts for the spent nuclear fuel shipments were estimated from data in Heiselmann (1994). The maximum number of spent nuclear fuel shipments of 691 per year would occur with Alternative 5b, Centralization at the INEL. A maximum 23-percent increase in traffic volume per day would occur with this alternative, based on the estimates of the number of trips required for the transport of construction equipment, material, spent nuclear fuel, other wastes, and workers to and from the INEL. Even if this average daily traffic volume were to occur for 1 hour, the maximum traffic volume would increase to 145 vehicles per hour for US 20, US 26, Routes 33 and 22; this would not change the baseline level of service, which is designated as "free flow."

5.11.5 Accident Impacts

An onsite spent nuclear fuel transportation accident involving the inadvertent shipment of a short-cooled fuel element from the Advanced Test Reactor to the Idaho Chemical Processing Plant was considered to be the maximum reasonably foreseeable accident. The melted spent nuclear fuel has potential to relocate into a critical configuration. However, the probability of a criticality accident is much less than 1×10^{-7} per year and would be considered to be not reasonably foreseeable. Table 5.11-1 lists the calculated maximally exposed individual dose and collective dose to general population in the maximally impacted sector and corresponding risk of fatal cancers. The dose to the maximally exposed individual is considered an occupational exposure.

As listed in Table 5.11-1, the total number of fatal cancers expected in the suburban population affected by the transportation for neutral and stable meteorological conditions would be 11 and 85, respectively. For the neutral case, this would represent a 0.01-percent increase from the number of fatal cancers that would be likely from normal incidence in the affected population. For the stable case, this would represent a 0.20-percent increase from the number of fatal cancers that would be likely from normal incidence in the affected population.

The total number of fatal cancers expected in the rural population affected by the transportation for neutral and stable meteorological conditions would be 0.75 and 6.0, respectively. For the neutral

Table 5.11-1. Impacts from maximum reasonably foreseeable spent nuclear fuel transportation accident on INEL[a] (using generic rural and suburban population densities).

Population density category[b]	Meteorology[c]	Accident frequency[d] (events/yr)	Dose to MEI[e] (rem)	Offsite population dose (person-rem)	Risk of fatal cancer per year[f]
Rural	Neutral	1.0×10^{-6}	7.6×10^{1}	1.5×10^{3}	7.5×10^{-7} (7.5×10^{-1})
Rural	Stable	1.0×10^{-7}	2.5×10^{2}	1.2×10^{4}	6.0×10^{-7} (6.0×10^{0})
Suburban	Neutral	1.0×10^{-6}	7.6×10^{1}	2.1×10^{4}	1.1×10^{-5} (1.1×10^{1})
Suburban	Stable	1.0×10^{-7}	2.5×10^{2}	1.7×10^{5}	8.5×10^{-6} (8.5×10^{1})

a. Source: Enyeart (1994).

b. Results are for generic rural and suburban population densities. The generic rural population density has an average population of 6 persons per square kilometer; the generic suburban population density has an average population of 719 persons per square kilometer. For comparison, the sector with the highest population density within 80 kilometers (50 miles) is due east of the Idaho Chemical Processing Plant and Test Reactor Area at the INEL with an average population density of 53 persons/km^2.

c. Neutral meteorology is characterized by Stability Class D, 4 meters-per-second wind speed, and occuring approximately 50 percent of the time. Stable meteorology is characterized by Stability Class F, 1 meter-per-second wind speed, and occuring approximately 5 percent of the time.

d. Accident frequency includes both the event frequency and the frequency of the meteorology. The frequency of stable meteorology is approximately one-tenth the frequency of neutral meteorology.

e. Maximally exposed individual located at the point of maximum exposure to the airborne release approximately 160 to 390 meters (525 to 1,280 feet) downwind, depending on meteorology. For onsite accidents the maximally exposed individual is assumed to be an INEL worker.

f. Fatal cancer risk = dose times accident frequency times (ICRP 60 risk factor for fatal cancers). The ICRP 60 risk factor is 5.0×10^{-4} fatal cancer per rem for public, 4.0×10^{-4} fatal cancer per rem for workers. For doses of 20 rem or more, the ICRP 60 conversion factor is doubled. Numbers in parentheses indicate the total number of fatal cancers in the population if the accident occurs. The maximally exposed individual dose is considered an occupational exposure.

case, this would represent a 0.09-percent increase from the number of fatal cancers that would be likely from normal incidences in the affected population. For the stable case, this would represent a 1.7-percent increase from the number of fatal cancers that would be likely from normal incidence in the affected population.

The estimated maximum nonradiological occupational and general population traffic fatalities over 40 years due to any of the spent nuclear fuel management alternatives would be 7.1×10^{-4} and 2.5×10^{-3}, respectively. These estimated fatalities were based on fatality risk factors for spent fuel shipments (Cashwell et. al 1986).

5.11.6 Onsite Mitigative and Preventative Measures

All onsite shipments would be in compliance with DOE ID Directive 5480.3, "Hazardous Materials Packaging and Transportation Safety Requirements." These requirements provide assurance that, under normal conditions, the INEL would meet as-low-as-reasonably-achievable conditions, reasonably foreseeable accident situations (those with a probability of occurrence greater than 1×10^{-7} per year) would not result in a loss of shielding or containment or a criticality, and an unintentional release of radioactive material would generate a timely response.

DOE would approve the type packages used for onsite shipments or would obtain a Nuclear Regulatory Commission or DOE certificate of compliance. If the Type B onsite package did not have Nuclear Regulatory Commission or DOE certification, the user of the package would have to establish how administrative controls and site-mitigating circumstances would ensure that the package would maintain containment and shielding integrity. The administrative and emergency response considerations would provide sufficient control so that accidents would not result in loss of containment or shielding, in criticality, or in an uncontrolled release of radioactive material that would create a hazard to the health and safety of the public or workers.

In the event of an accident, each DOE site has an established emergency management program. This program incorporates activities associated with emergency planning, preparedness, and response. Participating government agencies with plans that are interrelated with the INEL Emergency Plan for Action include the State of Idaho, Bingham County, Bonneville County, Butte County, Clark County, Jefferson County, the Bureau of Indian Affairs, and Fort Hall Indian Reservation. When an emergency condition exists at a facility, the Emergency Action Director is responsible for recognition, classification, notification, and protective action recommendations. At INEL emergency preparedness resources include fire protection, radiological and hazardous chemical material response, emergency control center, the INEL Warning Communication Center, the INEL Site Emergency Operational Center, and medical facilities.

5.12 Occupational and Public Health and Safety

This section presents DOE's estimates of the health effects from spent nuclear fuel-related activities at the INEL for the following human receptor groups:

- Involved Workers - workers at the facilities involved with spent nuclear fuel alternatives, including existing workers and new hires for selected alternative

- Maximally Exposed Individual (MEI) - person residing at the INEL site boundary

- Population - the general offsite population in the INEL region

- Construction Worker - labor force associated with construction activities

- Nonconstruction Worker - DOE labor force associated with nonconstruction activities

Radiological, chemical, and industrial safety hazards were considered in the estimates.

5.12.1 Radiological Exposure and Health Effects

The measure of impact used for evaluation of potential radiation exposures is risk of fatal cancers. Worker and maximally exposed individual effects are reported as individual radiation dose (in rem) and the estimated lifetime probability of fatal cancer. Population effects are reported as collective radiation dose (in person-rem) and the estimated number of fatal cancers in the affected population. Tables 5.12-1, 5.12-2, 5.12-3, and 5.12-4 summarize the radiological health effects calculations for each alternative.

Activities that workers would perform under each of the alternatives would be similar to those currently performed at the INEL. Therefore, the potential hazards encountered in the workplace would be similar to those that currently exist at the INEL. Further, DOE would mitigate these hazards with occupational and radiological safety programs operating under the same regulatory standards and limits that currently apply at the INEL. For these reasons, DOE anticipates that the average radiation dose

Table 5.12-1. Annual occupational radiation exposure and employment summary.[a]

	No Action (1)	Decentralization (2)	1992/1993 Planning Basis (3)	Regionalization by Fuel Type (4a)[b]	Centralization at Other DOE Sites (5a)	Centralization at the INEL (5b)
Number of Workers (annual average over years 1995-2004)[c]	1	1	200	200	10	200
Worker Collective Dose[d] (person-rem/year)	0.027	0.027	5.4	5.4	0.27	5.4

a. Source: Johnson (1995).
b. Alternative 4b(1), Regionalization by Geography (INEL), values are the same as those for Alternative 5b. Alternative 4b(2), Regionalization by Geography (Elsewhere), values are the same as those for Alternative 5a.
c. This 10-year average yields conservatively high employment; the 40-year average would be lower but data do not exist.
d. Based on thermoluminescence dosimetry records.

Table 5.12-2. Annual nonoccupational radiation exposure summary.

	No Action (1)	Decentralization (2)	1992/1993 Planning Basis (3)	Regionalization by Fuel Type (4a)[b]	Centralization at Other DOE Sites (5a)	Centralization at the INEL (5b)
MEI Dose (mrem/year)	3.5×10^{-3}	3.5×10^{-3}	8.0×10^{-3}	8.0×10^{-3}	3.9×10^{-3}	4.8×10^{-2}
Population Dose[a] (person-rem/year)	1.0×10^{-1}	1.0×10^{-1}	1.9×10^{-1}	1.9×10^{-1}	8.3×10^{-2}	3.9×10^{-1}

a. Population dose is calculated based on the projected population in 2000.
b. Alternative 4b(1), Regionalization by Geography (INEL), values are the same as those for Alternative 5b. Alternative 4b(2), Regionalization by Geography (Elsewhere), values are the same as those for Alternative 5a.

Table 5.12-3. Annual fatal cancer incidence and probability summary from radiological exposure.[a]

	No Action (1)	Decentralization (2)	1992/1993 Planning Basis (3)	Regionalization by Fuel Type(4a)[b]	Centralization at Other DOE Sites (5a)	Centralization at the INEL (5b)
Worker						
probability	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}
incidence	1×10^{-5}	1×10^{-5}	2×10^{-3}	2×10^{-3}	1×10^{-4}	2×10^{-3}
Maximally exposed member of the public probability	2×10^{-9}	2×10^{-9}	4×10^{-9}	4×10^{-9}	2×10^{-9}	2×10^{-8}
Population incidence	5×10^{-5}	5×10^{-5}	1×10^{-4}	1×10^{-4}	4×10^{-5}	2×10^{-4}

a. Risk factors for the worker (4×10^{-4} probability of occurrence per rem) or offsite population (5×10^{-4} probability of occurrence per rem) recommended by the International Commission on Radiological Protection (ICRP 1991).
b. Alternative 4b(1), Regionalization by Geography (INEL), values are the same as those for Alternative 5b. Alternative 4b(2), Regionalization by Geography (Elsewhere), values are the same as those for Alternative 5a.

Table 5.12-4. 40-year fatal cancer incidence summary from radiological exposure.[a]

	No Action (1)	Decentralization (2)	1992/1993 Planning Basis (3)	Regionalization by Fuel Type (4a)	Centralization at Other DOE Sites (5a)	Centralization at the INEL (5b)
Workers incidence	4×10^{-4}	4×10^{-4}	8×10^{-2}	8×10^{-2}	4×10^{-3}	8×10^{-2}
Population incidence	2×10^{-3}	2×10^{-3}	4×10^{-3}	4×10^{-3}	2×10^{-3}	8×10^{-3}

a. Alternative 4b(1), Regionalization by Geography (INEL), values are the same as those for Alternative 5b. Alternative 4b(2), Regionalization by Geography (Elsewhere), values are the same as those for Alternative 5a.

and the number of reportable cases of injury and illness would be proportional to the number of workers at the INEL under each alternative.

Table 5.12-1 lists involved worker doses based on an historic annual average dose of 27 mrem determined from thermoluminescent dosimeter data of workers involved in various INEL radiological work over the period 1987 to 1991 (see Appendix F of Volume 2). As mentioned above, the hazards associated with spent nuclear fuel activities are the same as the hazards associated with other INEL activities. Table 5.12-2 lists the exposure summaries for the maximally exposed individual and offsite population, based on radioactive emissions from normal operations and those resulting from startup of proposed facilities for the various alternatives. Note that population collective dose is higher than worker collective dose only under alternatives 1 and 2. For the alternatives, there is only 1 SNF worker averaged over 40 years. The nonoccupational population has more people to be exposed. When the worker population increases under Alternatives 3, 4, and 5, the worker dose becomes higher than the population dose. Section 5.7 presents the exposure information. Dose calculations are based on air emissions only, and not water pathways because none of the alternatives would involve the discharge of pollutants to surface waters or to the subsurface. Section 5.8 summarizes water quality.

Table 5.12-3 summarizes the fatal cancer incidence and probability for workers, maximally exposed individuals, and the offsite population based on the risk factors consistent with those recommended by the International Commission on Radiological Protection (ICRP 1991). For all alternatives, the probability of developing fatal cancer for any individual would be low, with the maximum value of 1×10^{-5} for the involved worker. The calculated incidence of fatal cancer for the total number of workers for each alternative and the offsite population would be less than 1.

Table 5.12-4 summarizes the 40-year projection of fatal cancer incidence associated with the worker and offsite populations. The highest involved worker and offsite population incidence, 0.1 and 0.01, respectively, would be associated with Alternative 5b.

Radiation doses associated with construction activities would be as low as reasonably achievable and no greater than 2 rem per year to any worker. Historical offsite doses associated with the INEL are summarized in the Idaho National Engineering Laboratory Historical Dose Evaluation (DOE 1991). The Centers for Disease Control and Prevention is conducting a more comprehensive reconstruction of doses from INEL operations.

5.12.2 Nonradiological Exposure and Health Effects

The air quality data listed in Tables 5.7-1 and 5.7-2 were used to evaluate health impacts associated with potential exposure to two compound classes, criteria pollutant and toxic. Table 5.7-1 lists five pollutant criteria and Table 5.7-2 lists six toxic air pollutant compounds. The toxic compounds were classified as noncarcinogens or carcinogens, consistent with EPA designations published in the Integrated Risk Information System (IRIS) data base. However, the IRIS data base does not include sufficient data to perform a quantitative inhalation cancer risk assessment.

Nonradiological health effects (hazard indices) for the INEL worker or maximally exposed individual were estimated by summing the ratios of the appropriate pollutant concentrations and their applicable standards presented in Table 5.7-1 and Table 5.7-2. Table 5.7-1 presents criteria pollutant concentrations at public access roads, which are the maximum of those calculated at the INEL site boundary, public access roads inside the INEL site boundary, and the Craters of the Moon Wilderness Area. The hazard index for the five criteria pollutants is less than 1 (0.2) for the workers or the maximally exposed individual, based on concentrations for the longest averaging times presented in Table 5.7-1. Table 5.7-2 presents toxic air pollutant concentrations at the public access roads, which are the maximum when compared with concentrations at the INEL site boundary and the Craters of the Moon Wilderness Area. The hazard index for the toxic air pollutants is also less than 1 (0.8) for the workers or the maximally exposed individual, based on concentrations with annual averaging time consideration. Accordingly, health effects are unlikely for either the criteria pollutants or the toxic air pollutants from spent nuclear fuel-related activities. The hazard index is not a statistical probability; therefore, it cannot be interpreted as such.

5.12.3 Industrial Safety

This section describes the following measures of impact for workplace hazards: (1) total reportable injuries and illness and (2) fatalities in the work force. This analysis considered injury and fatality rates for construction workers only since the alternatives do not result in incremental changes in operations employment. Table 5.12-5 lists the maximum annual number of projected injuries and illnesses and fatalities for construction workers by alternatives based on the maximum employment levels for any year between 1995-2035.

Table 5.12-5. Annual industrial safety health effects incidence summary.[a,b]

	No Action (1)	Decentralization (2)	1992/1993 Planning Basis (3)	Regionalization by Fuel Type (4a)[c]	Centralization at other DOE Sites (5a)	Centralization at the INEL (5b)
Construction workers						
Injury/illness	0	0	23	23	3	23
Fatality	0	0	<1	<1	<1	<1

a. 1988-1992 averages for occupational injury/illness and fatality rates for DOE and contractor employees.
b. Sources: DOE (1993b) and Section 5.3 of this appendix.
c. Alternative 4b(1) values are the same as those for Alternative 5b. Alternative 4b(2) values are the same as those for Alternative 5a.

5.13 Idaho National Engineering Laboratory Services

This section discusses the potential impacts from spent nuclear fuel management on utilities and energy at the INEL. It considers the consumption of water, electrical energy, fossil-based fuels, and wastewater discharge at the INEL site.

5.13.1 Construction

Table 5.13-1 summarizes estimates of annual requirements for electricity, water, wastewater, and diesel fuel for construction activities associated with each alternative and compares them to projected 1995 use levels for these resources. In general, the smallest increase in the demand for site services would result from Alternatives 4b(2) and 5a [Regionalization by Geography (Elsewhere) and Centralization at Other DOE Sites] and the largest increase would be associated with Alternatives 4b(1) and 5b [Regionalization by Geography (INEL) and Centralization at INEL].

Table 5.13-1. Estimated increase in annual electricity, water, wastewater treatment, and fuel requirements for construction activities associated with each alternative.

Service	Projected 1995 usage w/o Alternative	Estimated additional demand construction			
		Alternatives 1 and 2	Alternatives 3 and 4a	Alternatives 4b(1) and 5b	Alternatives 4b(2) and 5a
Electricity (MWH[a] per year)	208,000	71	150	2,100	10
Water (millions of liters per year)[b]	6,450	No increase	2.1	2.2	0.5
Sanitary wastewater (millions of liters per year)	540	No increase	1.5	4.5	0.5
Diesel fuel (liters per year)	5,830,000	6,400	8,500	14,000	1,500

a. MWH = megawatt hours.
b. To convert liters to gallons, multiply by 0.264.
Source: Hendrickson (1995).

Under Alternatives 4b(1) and 5b, the estimated annual increases in utility and energy usage rates from construction activities would be 2,100 megawatt-hours of electricity, 2.2 million liters (580,000 gallons) of water, 4.5 million liters (1,200,000 gallons) of wastewater discharge, and 14,000 liters (3,700 gallons) of diesel fuel. These changes represent modest increases ranging from near zero percent to 1.0 percent above projected 1995 usage levels and are well within current system capabilities and usage limits (see Section 4.13). The other alternatives would result in smaller increases in energy usage and would have no adverse impact on utility services at the INEL.

5.13.2 Operations

Table 5.13-2 summarizes estimates of annual requirements for electricity, water, wastewater, and fuel for operations activities associated with each alternative and compares them to project 1995 INEL usage of these resources. In general, the smallest increase in the demand for site services would result from Alternatives 1 and 2 (No-Action and Decentralization) and the largest would be associated with Alternatives 4b(1) and 5b [Regionalization by Geography (INEL) and Centralization at INEL].

Table 5.13-2. Estimated increase in annual electricity, water, wastewater treatment, and fuel requirements for operations activities associated with each alternative.

Service	Projected 1995 usage w/o Alternative	Estimated additional demand operation			
		Alternatives 1 and 2	Alternatives 3 and 4a	Alternatives 4b(1) and 5b	Alternatives 4b(2) and 5a
Electricity (MWH[a] per year)	208,000	180	2,200	11,000	2,000
Water (millions of liters per year)[b]	6,450	No increase	No increase	48	No increase
Sanitary wastewater (millions of liters per year)[c]	540	No increase	No increase	0.3	No increase
Fuel oil (liters per year)	11,100,000	28,000	330,000	1,100,000	300,000

a. MWH = megawatt hours.
b. To convert liters to gallons, multiply by 0.264.
c. Some industrial wastewater, such as steam condensate, is also discharged to evaporation ponds and injection wells.
Sources: Hendrickson (1995).

Under Alternatives 4b(1) and 5b, the estimated annual increases in utility and energy usage rates from operations activities would be 11,000 megawatt-hours of electricity, 48 million liters (13 million gallons) of water, 0.3 million liters (79,000 gallons) of wastewater, and 1,100,000 liters (290,000 gallons) of fuel oil. These changes represent modest increases ranging from near zero percent to 10 percent and are well within current system capabilities and usage limits (see Section 4.13). The other alternatives would result in smaller increases in energy usage and would have no adverse impact on utility services at the INEL.

5.14 Materials and Waste Management

This section discusses the impacts to the management of materials and wastes at the INEL site and Idaho Falls facilities as a result of the implementation of the spent nuclear fuel management alternatives. Alternatives 4b(1), and 5b, both with the spent fuel processing option, each establish the upper bound of potential impacts on projected rates of generation, treatment, storage, and disposal inventories of materials and wastes. Table 5.14-1 and 5.14-2 summarize waste generation projections for each alternative. The tables present average generating rates over the life cycle of each alternative and maximum annual increments over peak generation periods.

5.14.1 Alternative 1 - No Action

Under the No Action Alternative, 9 cubic meters of industrial solid waste would be generated during construction of the Alternate Fuel Storage Facility for the TAN Pool Fuel Transfer Project at the Idaho Chemical Processing Plant. At the completion of this project in 1998, there would be 485 cubic meters of non-fuel solid low-level waste consisting of Three Mile Island hardware and metals that would be removed and dispositioned in a separate project. These impacts apply also to the description of impacts for the other spent nuclear fuel management alternatives with the exception of Alternatives 4b(2) and 5a. The non-fuel solid low-level waste is already existing; therefore, it is not included in Table 5.14-1 as an increase in low-level waste generation.

5.14.2 Alternative 2 - Decentralization

In general, the character of the impacts to materials and waste management would be similar to those under the No Action Alternative.

5.14.3 Alternative 3 - 1992/1993 Planning Basis

Industrial solid waste would be generated from construction and operation of the various SNF projects under Alternative 3. This nonradioactive waste would be disposed of in the Central Facilities Area landfill. Landfill space is nonrestrictive for industrial solid waste disposal. Construction phase activities would generate a cumulative total of 620 cubic meters of industrial and commercial solid

Table 5.14-1. Average annual waste generation projections for selected SNF management alternatives at INEL.[a]

Alternative	Waste type	Average annual increment over 1995 baseline			
		Phase	Period (years)	Increase (percent)	Annual rate (cubic meters per year)
No Action (Alternative 1) and Decentralization (Alternative 2)	Industrial	Construction	1995-1996	0.02	9
1992/1993 Planning Basis (Alternative 3) and Regionalization by Fuel Type (Alternative 4a)	Industrial	Construction	1995-2005	0.1	62
		Operation	1996-2035	1.2	600
	Low-Level[b,c]	Construction	1995-1999	8.6	370
		Operation	1996-2035	4.6	200
	High-Level	Operation	1996-2024	0.1	3
	Mixed Low-Level	Operation	1996-2024	<0.1	<1
	Transuranic	Operation	1996-2024	530	32
Regionalization by Geography (INEL) [Alternative 4b(1)] and Centralization at INEL (Alternative 5b)	Industrial	Construction	1995-2008	0.6	290
		Operation	1996-2035	5.0	2,600
	Low-Level[b,c]	Construction	1995-1999	8.6	370
		Operation	1996-2035	9.6	410
	High-Level	Operation	1996-2035	15.7	120
	Mixed Low-Level	Operation	1996-2024	<0.1	<1
	Transuranic	Operation	1996-2024	530	32
Regionalization by Geography (Elsewhere) [Alternative 4b(2)] and Centralization at Other DOE Sites (Alternative 5a)	Industrial	Construction	1995-1996	<0.1	50
		Operation	1996-2024	0.4	210
	Low-Level	Operation	1996-2024	1.9	83
	High-Level	Operation	1996-2024	0.1	3
	Mixed Low-Level	Operation	1996-2024	<0.1	<1
	Transuranic	Operation	1996-2024	530	32

a. Source: Appendix C of Volume 2 of this EIS.
b. Low-level waste from TAN Pool Fuel Transfer Project to be removed and dispositioned in a separate project not included for any alternatives.
c. Low-level waste generated from dispositioning and decontamination of fuel racks not included in any alternatives.

Table 5.14-2. Peak waste generation highlights for selected SNF management alternatives at INEL.[a]

		Maximum increment over 1995 baseline			
Alternative	Waste type	Phase	Period (years)	Increase (percent)	Annual rate (cubic meters per year)
No Action (Alternative 1) and Decentralization (Alternative 2)	Industrial	Construction	1995-1996	0.02	9
1992/1993 Planning Basis (Alternative 3) and Regionalization by Fuel Type (Alternative 4a)	Industrial	Construction	1995-1996	0.4	220
		Operation	2005-2021	1.6	810
	Low-Level[b,c]	Construction	1995-1997	13.4	570
		Operation	2005-2024	6.1	260
		Concurrent Activity[d]	1996-1997	14.2	610
	High-Level	Operation	1997-1998	0.2	6
	Mixed Low-Level	Operation	1997-1998	<0.1	<1
	Transuranic	Operation	1997-1998	600	36
Regionalization by Geography (INEL) [Alternative 4b(1)] and Centralization at INEL (Alternative 5b)	Industrial	Construction	1999-2006	0.9	450
		Operation	2008-2021	6.8	3,500
	Low-Level[b,c]	Construction	1995-1997	13.4	570
		Operation	2008-2024	13.3	570
		Concurrent Activity[d]	1996-1997	14.2	610
	High-Level	Operation	2005-2024	21.1	160
	Mixed Low-Level	Operation	1997-1998	<0.1	<1
	Transuranic	Operation	1997-1998	600	36
Regionalization by Geography (Elsewhere) [Alternative 4b(2)] and Centralization at Other DOE Sites (Alternative 5a)	Industrial	Construction	1995-1996	<0.1	50
		Operation	1996-2024	0.4	210
	Low-Level	Operation	1996-2010	3.1	130
	High-Level	Operation	1996-2024	0.1	3
	Mixed Low-Level	Operation	1996-2024	<0.1	<1
	Transuranic	Operation	1996-2024	530	32

a. Source: Appendix C of Volume 2 of this EIS.
b. Low-level waste from TAN Pool Fuel Transfer Project to be removed and dispositioned in a separate project not included for any alternatives.
c. Low-level waste generated from dispositioning and decontamination of fuel racks not included in any alternatives.
d. Construction and operations occurring simultaneously.

waste. The Fuel Receiving, Canning, Characterization, and Shipping Facility will generate the most industrial waste of any of the projects, 490 cubic meters per year from 2005 through 2035.

In addition, the Fuel Receiving, Canning, Characterization, and Shipping Facility will generate 220 cubic meters per year of low-level waste during the same period. The Dry Storage Facility would generate an additional 5 cubic meters of low-level waste annually from 2005 through 2035. Including liquid low-level waste, the Increased Rack Capacity and Additional Increased Rack Capacity projects would increase generation rates by 570 cubic meters annually during construction from 1995 through 1997. Low-level waste would decrease to approximately 160 cubic meters per year from 1997 through 1999 with the completion of the Increased Rack Capacity project. Liquid low-level waste would be disposed in existing liquid waste processing systems at the Idaho Chemical Processing Plant. Solid radioactive wastes would be packaged and disposed of at the Radioactive Waste Management Complex, or incinerated at the Waste Experimental Reduction Facility, whichever is appropriate. Low-level waste from reracking fuel racks for the Increased Rack Capacity Project will be decontaminated and dispositioned by a licensed commercial vendor.

Experimental Breeder Reactor-II Blanket Treatment will generate 7 cubic meters of low-level waste for 1 year from 1997 to 1998.

The storage of low-level waste for incineration is not considered to be restrictive between 1995 through 2005. However, beyond 2005, low-level waste storage capacity may become strained. Use of commercial facilities to incinerate the backlog of low-level waste is under consideration in order to reduce or prevent the accumulation of low-level waste, but no firm commitment or contract has yet been established (EG&G 1993a).

The Radioactive Waste Management Complex appears to have adequate disposal capacity for low-level waste between 1995 and 2005. However, beyond 2005, additional capacity may be required. Excess capacity would be provided with the development of the proposed Low-Level Waste/Mixed Low-Level Waste Disposal Facility (EG&G 1993a).

The Electrometallurgical Process Demonstration Project will generate high-level, mixed low-level, low-level, transuranic, and industrial wastes from the demonstration and testing of new spent fuel management processes from 1996 through 2024.

Experimental Breeder Reactor-II Blanket Treatment will also generate high-level, mixed low-level, and transuranic wastes.

High-level waste would be immobilized after 2005, and may eventually be transported to a Federal high-level waste and spent nuclear fuel repository for disposal. Transuranic waste meeting waste acceptance criteria to be developed could be shipped to a potential Federal repository for disposal should one be selected (EG&G 1993a).

5.14.4 Alternative 4a - Regionalization by Fuel Type

In general, the character of the impacts to materials and waste management would be similar to those under Alternative 3.

5.14.5 Alternative 4b(1) - Regionalization by Geography (INEL)

The character and intensity of impacts on waste management activities at the INEL are similar to those under Alternatives 3 and 4a for some of the SNF management projects including the TAN Pool Fuel Transfer Project at the Idaho Chemical Processing Plant; the Increased Rack Capacity and Additional Increased Rack Capacity projects; the Experimental Breeder Reactor-II Blanket Treatment facility; and the Electrometallurgical Process Demonstration Project. Under Alternative 4b(1), the Dry Fuel Storage Facility is expanded and Fuel Receiving, Canning/Characterization, and Shipping Facility waste streams decrease relative to Alternatives 3 and 4a; however, the net effect of these differences on industrial/commercial solid waste generation and low-level waste generation for both construction and operation results in waste generation rates similar to those under Alternatives 3 and 4a.

The increase in average and peak generation rates over Alternatives 3 and 4a (Tables 5.14-1 and 5.14-2) is due to the Spent Fuel Processing option included under Alternative 4b(1), which accounts for the relative increase in generation rates over Alternatives 3 and 4a. Fuel processing would be done in order to stabilize the spent nuclear fuel and remove risks associated with storage and disposal, and to manage the resultant high-level waste in a cost-effective manner. If this alternative were pursued aggressively, the generated high-level waste residual resulting from segregating fissile material from the spent nuclear fuel may require additional high-level waste tankage. This increase in capacity would be covered by the High-Level Tank Farm New Tanks project described in Volume 2 of the EIS.

Capacity discussions for industrial/commercial solid waste and low-level waste under Alternative 3 apply to Alternative 4b(1).

5.14.6 Alternative 4b(2) - Regionalization by Geography (Elsewhere)

Construction phase activities would generate a cumulative total of 50 cubic meters of industrial and commercial solid waste. Overall, waste generation would be lower than all of the SNF management alternatives, with the exceptions of the No Action and Decentralization Alternatives.

5.14.7 Alternative 5a - Centralization at Other DOE Sites

In general, the character of the impacts to materials and waste management would be similar to those under Alternative 4b(2).

5.14.8 Alternative 5b - Centralization at the INEL

In general, the character of the impacts to materials and waste management would be similar to those under Alternative 4b(1).

5.15 Accidents

5.15.1 Introduction

Activities associated with the transportation, receipt, handling, stabilization, and storage of spent nuclear fuel at the INEL involve substantial quantities of radioactive materials and limited quantities of toxic chemicals. Under certain circumstances, the potential exists for accidents involving these materials to occur, which would result in exposure to INEL workers or members of the public, or contamination of the surrounding environment. Accidents can be categorized as follows:

- Abnormal events such as minor spills
- Design-basis events, which a facility is designed to withstand
- Beyond-design-basis events, which a facility is not designed to withstand (but whose consequences it may nevertheless mitigate)

This section summarizes postulated radiological and toxic material accidents in each accident category and describes their estimated consequences to workers, members of the public, and the environment. The scope of this section is limited to accidents within facilities; transportation accidents between facilities are addressed in Section 5.11. [Further information on the accidents summarized in this section, as well as information on other "lower consequence" accidents analyzed, is provided in Slaughterbeck et al. (1995)].

An accident is a series of unexpected or undesirable "initiating" events that lead to a release of radioactive or toxic materials within a facility or to the environment. This analysis defines initiating events that can lead to a spent nuclear fuel-related facility accident in three broad categories: external initiators, internal initiators, and natural phenomena initiators. External initiators (e.g., aircraft crashes, and nearby explosions or toxic material releases) originate outside the facility and can affect the ability of the facility to maintain confinement of radioactive or hazardous material. Internal initiators originate within a facility (e.g., equipment failures or human error) and are usually the result of facility operation. Sabotage and terrorist activities (i.e., intentional human initiators) might be either external or internal initiators. Natural phenomena initiators include weather-related (e.g., floods and tornadoes) and seismic events. This analysis defines initiators in terms of events that cause, directly or indirectly, a release of radioactive or hazardous materials within a facility or to the environment by failure or bypass of confinement.

Tables 5.15-1 through 5.15-4 summarize the radiological results of the analyses described in this section. Section 5.15.2 summarizes historic accidents at the INEL associated with spent nuclear fuel-related activities. Section 5.15.3 describes the methodology used to identify and evaluate potential radiological accidents associated with spent nuclear fuel receipt, handling, storage, and intra-area transportation activities. Sections 5.15.4 and 5.15.5 evaluate the postulated maximum reasonably foreseeable radiological and toxic material accidents, respectively.

5.15.2 Historic Perspective

Many of the actions proposed under the different spent nuclear fuel management alternatives considered in this EIS are continuations or variations of past practices at the INEL. DOE has analyzed consequences to the public from historic INEL accidents in detail and has determined them to be low (DOE 1991).

Consequences of accidents can involve fatalities, injuries, or illness. Fatalities can be prompt (immediate), such as in construction accidents, or latent (delayed), such as cancer caused by radiation exposure. While public comments received in scoping meetings for this EIS included many concerns about potential accidents at the INEL, the historic record demonstrates that DOE facilities, including the INEL, have a very good safety record, particularly in comparison to commercial industries (e.g., agriculture and construction). Figure 5.15-1 shows the rate of worker fatalities at the INEL and other DOE sites (DOE 1993b) compared to national-average rates that the National Safety Council compiled over a 10-year period for various industry groups (NSC 1993) and State of Idaho average rates (Hendrix 1994). While past accident occurrence rates are not necessarily indicative of future rates, the historic record reflects the DOE emphasis on safe operations.

There have been no prompt fatalities and no known latent fatalities to members of the public from accidental releases of radioactive or hazardous materials associated with spent nuclear fuel management activities in the 40-year history of INEL facilities, although some accidents associated

Table 5.15-1. Summary of radiological accidents for worker located 100 meters downwind from the point of release.

Accident Description	Attribute	Alternative 1 No Action	Alternative 2 Decentralization	Alternative 3 1992/1993 Planning Basis	Alternative 4a[a] Regionalization by Fuel Type	Alternative 5a Centralization at Other Sites	Alternative 5b Centralization at the INEL
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[b]	Consequences[c]	(d)	(d)	(d)	(d)	(d)	(d)
	Adjusted annual frequency	1.0×10^{-2}	1.2×10^{-2}	3.1×10^{-2}	4.8×10^{-2}	8.6×10^{-2}	2.0×10^{-1}
	Adjusted point estimate of risk[e]	(d)	(d)	(d)	(d)	(d)	(d)
2. Uncontrolled chain reaction (criticality) at ICPP[f]	Consequences[c]	3.9×10^{-5}	3.9×10^{-5}	3.9×10^{-5}	3.9×10^{-5}	3.9×10^{-5}	3.9×10^{-5}
	Adjusted annual frequency	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}
	Adjusted point estimate of risk[e]	4.0×10^{-8}	4.0×10^{-8}	4.0×10^{-8}	4.0×10^{-8}	4.0×10^{-8}	4.0×10^{-8}
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	Consequences[c]	2.5×10^{-4}	2.5×10^{-4}	2.5×10^{-4}	2.5×10^{-4}	2.5×10^{-4}	2.5×10^{-4}
	Adjusted annual frequency	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}
	Adjusted point estimate of risk[e]	2.5×10^{-9}	2.5×10^{-9}	2.5×10^{-9}	2.5×10^{-9}	2.5×10^{-9}	2.5×10^{-9}
4. Material release from HFEF resulting from aircraft crash and ensuing fire	Consequences[c]	1.8×10^{-3}	1.8×10^{-3}	1.8×10^{-3}	1.8×10^{-3}	1.8×10^{-3}	1.8×10^{-3}
	Adjusted annual frequency	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]
	Adjusted point estimate of risk[e]	1.8×10^{-10}	1.8×10^{-10}	1.8×10^{-10}	1.8×10^{-10}	1.8×10^{-10}	1.8×10^{-10}
5. Inadvertent nuclear criticality at ICPP[f] CPP-666 during processing	Consequences[c]	(h)	(h)	(h)	(h)	(h)	3.6×10^{-3}
	Adjusted annual frequency	(h)	(h)	(h)	(h)	(h)	1.0×10^{-3}
	Adjusted point estimate of risk[e]	(h)	(h)	(h)	(h)	(h)	3.6×10^{-6}
6. Hydrogen explosion in ICPP[f] CPP-666 dissolver	Consequences[c]	(h)	(h)	(h)	(h)	(h)	(d)
	Adjusted annual frequency	(h)	(h)	(h)	(h)	(h)	(d)
	Adjusted point estimate of risk[e]	(h)	(h)	(h)	(h)	(h)	(d)

Table 5.15-1. (continued).

Accident Description	Attribute	Alternative 1 No Action	Alternative 2 Decentralization	Alternative 3 1992/1993 Planning Basis	Alternative 4a[a] Regionalization by Fuel Type	Alternative 5a Centralization at Other Sites	Alternative 5b Centralization at the INEL
7. Inadvertent dissolution of 30-day cooled fuel at ICPP[f] CPP-666	Consequences[c]	(h)	(h)	(h)	(h)	(h)	(d)
	Adjusted annual frequency	(h)	(h)	(h)	(h)	(h)	(d)
	Adjusted point estimate of risk[e]	(h)	(h)	(h)	(h)	(h)	(d)

a. The radiological accident results for Alternative 4b(1), "Regionalization by Geography (INEL)," are conservatively assumed to be the same as those presented for Alternative 5b, as discussed in Section 5.15.4.4. The radiological accident results for Alternative 4b(2), "Regionalization by Geography (Elsewhere)," are identical to those presented for Alternative 5a, as discussed in Section 5.15.4.4.

b. HFEF = Hot Fuel Examination Facility.

c. Consequences are presented in terms of latent fatal cancers based on conservative (95 percentile) meteorological conditions. Consequences are calculated by multiplying the estimated exposure (i.e., dose) by an International Commission on Radiological Protection conversion factor of 4.0×10^{-4} cancer per rem for an adult worker (or 8.0×10^{-4} cancer per rem if the estimated exposure is greater than 20 rem).

d. The safety analysis report utilized for this accident analysis does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. As demonstrated by the dose to the maximally exposed individual, consequences to the public from Accident 1 could be less than the consequences from Accidents 2 through 4. However, given the high frequency for Accident 1 compared to Accidents 2 through 4, the risk could actually be greater than for Accidents 2 through 4.

e. This attribute is equal to consequences × frequency (events per year). The information is based on conservative (95 percentile) meteorological conditions.

f. ICPP = Idaho Chemical Processing Plant.

g. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

h. Resuming processing at the INEL under this alternative is not considered.

Table 5.15-2. Summary of radiological accidents for individual located at the nearest point of public access within the site boundary.

Accident Description	Attribute	Alternative 1 No Action	Alternative 2 Decentralization	Alternative 3 1992/1993 Planning Basis	Alternative 4a[a] Regionalization by Fuel Type	Alternative 5a Centralization at Other Sites	Alternative 5b Centralization at the INEL
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[b]	Consequences[c]	(d)	(d)	(d)	(d)	(d)	(d)
	Adjusted annual frequency	1.0×10^{-2}	1.2×10^{-2}	3.1×10^{-2}	4.8×10^{-2}	8.6×10^{-2}	2.0×10^{-1}
	Adjusted point estimate of risk[e]	(d)	(d)	(d)	(d)	(d)	(d)
2. Uncontrolled chain reaction (criticality) at ICPP[f]	Consequences[c]	7.0×10^{-7}	7.0×10^{-7}	7.0×10^{-7}	7.0×10^{-7}	7.0×10^{-7}	7.0×10^{-7}
	Adjusted annual frequency	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}
	Adjusted point estimate of risk[e]	7.0×10^{-10}	7.0×10^{-10}	7.0×10^{-10}	7.0×10^{-10}	7.0×10^{-10}	7.0×10^{-10}
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	Consequences[c]	3.3×10^{-4}	3.3×10^{-4}	3.3×10^{-4}	3.3×10^{-4}	3.3×10^{-4}	3.3×10^{-4}
	Adjusted annual frequency	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}
	Adjusted point estimate of risk[e]	3.3×10^{-9}	3.3×10^{-9}	3.3×10^{-9}	3.3×10^{-9}	3.3×10^{-9}	3.3×10^{-9}
4. Material release from HFEF resulting from aircraft crash and ensuing fire	Consequences[c]	1.6×10^{-4}	1.6×10^{-4}	1.6×10^{-4}	1.6×10^{-4}	1.6×10^{-4}	1.6×10^{-4}
	Adjusted annual frequency	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]
	Adjusted point estimate of risk[e]	1.6×10^{-11}	1.6×10^{-11}	1.6×10^{-11}	1.6×10^{-11}	1.6×10^{-11}	1.6×10^{-11}
5. Inadvertent nuclear criticality ICPP[f] CPP-666 during processing	Consequences[c]	(h)	(h)	(h)	(h)	(h)	2.5×10^{-5}
	Adjusted annual frequency	(h)	(h)	(h)	(h)	(h)	1.0×10^{-3}
	Adjusted point estimate of risk[e]	(h)	(h)	(h)	(h)	(h)	2.5×10^{-8}
6. Hydrogen explosion in ICPP[f] CPP-666 dissolver	Consequences[c]	(h)	(h)	(h)	(h)	(h)	(d)
	Adjusted annual frequency	(h)	(h)	(h)	(h)	(h)	(d)
	Adjusted point estimate of risk[e]	(h)	(h)	(h)	(h)	(h)	(d)

Table 5.15-2. (continued).

Accident Description	Attribute	Alternative 1 No Action	Alternative 2 Decentralization	Alternative 3 1992/1993 Planning Basis	Alternative 4a[a] Regionalization by Fuel Type	Alternative 5a Centralization at Other Sites	Alternative 5b Centralization at the INEL
7. Inadvertent dissolution of 30-day cooled fuel at ICPP[f] CPP-666	Consequences[c]	(h)	(h)	(h)	(h)	(h)	(d)
	Adjusted annual frequency	(h)	(h)	(h)	(h)	(h)	(d)
	Adjusted point estimate of risk[e]	(h)	(h)	(h)	(h)	(h)	(d)

a. The radiological accident results for Alternative 4b(1), "Regionalization by Geography (INEL)," are conservatively assumed to be the same as those presented for Alternative 5b, as discussed in Section 5.15.4.4. The radiological accident results for Alternative 4b(2), "Regionalization by Geography (Elsewhere)," are identical to those presented for Alternative 5a, as discussed in Section 5.15.4.4.

b. HFEF = Hot Fuel Examination Facility.

c. Consequences are presented in terms of latent fatal cancers based on conservative (95 percentile) meteorological conditions. Consequences are calculated by multiplying the estimated exposure (i.e., dose) by an International Commission on Radiological Protection conversion factor of 5.0×10^{-4} cancer per person-rem for the offsite population (or 1.0×10^{-3} cancer per rem if the estimated population exposure is greater than 20 rem for any individual member of the public).

d. The safety analysis report utilized for this accident analysis does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. As demonstrated by the dose to the maximally exposed individual, consequences to the public from this accident could be less than the consequences from Accidents 2 through 4. However, given the high frequency for this accident compared to Accidents 2 through 4, the risk could actually be greater than for Accidents 2 through 4.

e. This attribute is equal to consequences × frequency (events per year). The information is based on conservative (95 percentile) meteorological conditions.

f. ICPP = Idaho Chemical Processing Plant.

g. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

h. Resuming processing at the INEL under this alternative is not considered.

Table 5.15-3. Summary of radiological accidents for maximally exposed hypothetical individual located at the nearest site boundary.

Accident Description	Attribute	Alternative 1 No Action	Alternative 2 Decentralization	Alternative 3 1992/1993 Planning Basis	Alternative 4a[a] Regionalization by Fuel Type	Alternative 5a Centralization at Other Sites	Alternative 5b Centralization at the INEL
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[b]	Consequences[c]	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}
	Adjusted annual frequency	1.0×10^{-2}	1.2×10^{-2}	3.1×10^{-2}	4.8×10^{-2}	8.6×10^{-2}	2.0×10^{-1}
	Adjusted point estimate of risk[d]	1.0×10^{-8}	1.2×10^{-8}	3.1×10^{-8}	4.8×10^{-8}	8.6×10^{-8}	2.0×10^{-7}
2. Uncontrolled chain reaction (criticality) at ICPP[e]	Consequences[c]	5.0×10^{-7}	5.0×10^{-7}	5.0×10^{-7}	5.0×10^{-7}	5.0×10^{-7}	5.0×10^{-7}
	Adjusted annual frequency	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}
	Adjusted point estimate of risk[d]	5.0×10^{-10}	5.0×10^{-10}	5.0×10^{-10}	5.0×10^{-10}	5.0×10^{-10}	5.0×10^{-10}
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	Consequences[c]	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}
	Adjusted annual frequency	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}
	Adjusted point estimate of risk[d]	2.5×10^{-8}	2.5×10^{-8}	2.5×10^{-8}	2.5×10^{-8}	2.5×10^{-8}	2.5×10^{-8}
4. Material release from HFEF resulting from aircraft crash and ensuing fire	Consequences[c]	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}
	Adjusted annual frequency	1.0×10^{-7}[f]	1.0×10^{-7}[f]	1.0×10^{-7}[f]	1.0×10^{-7}[f]	1.0×10^{-7}[f]	1.0×10^{-7}[f]
	Adjusted point estimate of risk[d]	2.5×10^{-10}	2.5×10^{-10}	2.5×10^{-10}	2.5×10^{-10}	2.5×10^{-10}	2.5×10^{-10}
5. Inadvertent nuclear criticality ICPP[e] CPP-666 during processing	Consequences[c]	(g)	(g)	(g)	(g)	(g)	1.4×10^{-5}
	Adjusted annual frequency	(g)	(g)	(g)	(g)	(g)	1.0×10^{-3}
	Adjusted point estimate of risk[d]	(g)	(g)	(g)	(g)	(g)	1.4×10^{-8}
6. Hydrogen explosion in ICPP[e] CPP-666 dissolver	Consequences[c]	(g)	(g)	(g)	(g)	(g)	3.2×10^{-7}
	Adjusted annual frequency	(g)	(g)	(g)	(g)	(g)	1.0×10^{-5}
	Adjusted point estimate of risk[d]	(g)	(g)	(g)	(g)	(g)	3.2×10^{-12}

Table 5.15-3. (continued).

Accident Description	Attribute	Alternative 1 No Action	Alternative 2 Decentralization	Alternative 3 1992/1993 Planning Basis	Alternative 4a[a] Regionalization by Fuel Type	Alternative 5a Centralization at Other Sites	Alternative 5b Centralization at the INEL
7. Inadvertent dissolution of 30-day cooled fuel at ICPP[e] CPP-666	Consequences[c]	(g)	(g)	(g)	(g)	(g)	1.5×10^{-5}
	Adjusted annual frequency	(g)	(g)	(g)	(g)	(g)	1.0×10^{-6}
	Adjusted point estimate of risk[d]	(g)	(g)	(g)	(g)	(g)	1.5×10^{-11}

a. The radiological accident results for Alternative 4b(1), "Regionalization by Geography (INEL)," are conservatively assumed to be the same as those presented for Alternative 5b, as discussed in Section 5.15.4.4. The radiological accident results for Alternative 4b(2), "Regionalization by Geography (Elsewhere)," are identical to those presented for Alternative 5a, as discussed in Section 5.15.4.4.

b. HFEF = Hot Fuel Examination Facility.

c. Consequences are presented in terms of latent fatal cancers based on conservative (95 percentile) meteorological conditions. Consequences are calculated by multiplying the estimated exposure (i.e., dose) by an International Commission on Radiological Protection conversion factor of 5.0×10^{-4} cancer per person-rem for the offsite population (or 1.0×10^{-3} cancer per rem if the estimated population exposure is greater than 20 rem for any individual member of the public).

d. This is equal to consequences x frequency (events per year). The information is based on conservative (95 percentile) meteorological conditions.

e. ICPP = Idaho Chemical Processing Plant.

f. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

g. Resuming processing at the INEL under this alternative is not considered.

Table 5.15-4. Summary of radiological accidents for offsite population within 80 kilometers (50 miles) from the point of release.

Accident Description	Attribute	Alternative 1 No Action	Alternative 2 Decentralization	Alternative 3 1992/1993 Planning Basis	Alternative 4a[a] Regionalization by Fuel Type	Alternative 5a Centralization at Other Sites	Alternative 5b Centralization at the INEL
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[b]	Consequences[c]	(d)	(d)	(d)	(d)	(d)	(d)
	Adjusted annual frequency	1.0×10^{-2}	1.2×10^{-2}	3.1×10^{-2}	4.8×10^{-2}	8.6×10^{-2}	2.0×10^{-1}
	Adjusted point estimate of risk[e]	(d)	(d)	(d)	(d)	(d)	(d)
2. Uncontrolled chain reaction (criticality) at ICPP[f]	Consequences[c]	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}
	Adjusted annual frequency	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}
	Adjusted point estimate of risk[e]	3.0×10^{-7}	3.0×10^{-7}	3.0×10^{-7}	3.0×10^{-7}	3.0×10^{-7}	3.0×10^{-7}
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	Consequences[c]	7.0×10^{0}	7.0×10^{0}	7.0×10^{0}	7.0×10^{0}	7.0×10^{0}	7.0×10^{0}
	Adjusted annual frequency	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}
	Adjusted point estimate of risk[e]	7.0×10^{-5}	7.0×10^{-5}	7.0×10^{-5}	7.0×10^{-5}	7.0×10^{-5}	7.0×10^{-5}
4. Material release from HFEF resulting from aircraft crash and ensuing fire	Consequences[c]	1.0×10^{0}	1.0×10^{0}	1.0×10^{0}	1.0×10^{0}	1.0×10^{0}	1.0×10^{0}
	Adjusted annual frequency	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]	1.0×10^{-7}[g]
	Adjusted point estimate of risk[e]	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}
5. Inadvertent nuclear criticality ICPP[f] CPP-666 during processing	Consequences[c]	(h)	(h)	(h)	(h)	(h)	2.8×10^{-3}
	Adjusted annual frequency	(h)	(h)	(h)	(h)	(h)	1.0×10^{-3}
	Adjusted point estimate of risk[e]	(h)	(h)	(h)	(h)	(h)	2.8×10^{-6}
6. Hydrogen explosion in ICPP[f] CPP-666 dissolver	Consequences[c]	(h)	(h)	(h)	(h)	(h)	4.1×10^{-4}
	Adjusted annual frequency	(h)	(h)	(h)	(h)	(h)	1.0×10^{-5}
	Adjusted point estimate of risk[e]	(h)	(h)	(h)	(h)	(h)	4.1×10^{-9}

Table 5.15-4. (continued).

Accident Description	Attribute	Alternative 1 No Action	Alternative 2 Decentralization	Alternative 3 1992/1993 Planning Basis	Alternative 4a[a] Regionalization by Fuel Type	Alternative 5a Centralization at Other Sites	Alternative 5b Centralization at the INEL
7. Inadvertent dissolution of 30-day cooled fuel at ICPP[f] CPP-666	Consequences[c]	(h)	(h)	(h)	(h)	(h)	1.5×10^{-2}
	Adjusted annual frequency	(h)	(h)	(h)	(h)	(h)	1.0×10^{-6}
	Adjusted point estimate of risk[e]	(h)	(h)	(h)	(h)	(h)	1.5×10^{-8}

a. The radiological accident results for Alternative 4b(1), "Regionalization by Geography (INEL)," are conservatively assumed to be the same as those presented for Alternative 5b, as discussed in Section 5.15.4.4. The radiological accident results for Alternative 4b(2), "Regionalization by Geography (Elsewhere)," are identical to those presented for Alternative 5a, as discussed in Section 5.15.4.4.

b. HFEF = Hot Fuel Examination Facility.

c. Consequences are presented in terms of latent fatal cancers based on conservative (95 percentile) meteorological conditions. Consequences are calculated by multiplying the estimated exposure (i.e., dose) by an International Commission on Radiological Protection conversion factor of 5.0×10^{-4} cancer per person-rem for the offsite population (or 1.0×10^{-3} cancer per rem if the estimated population exposure is greater than 20 rem for any individual member of the public).

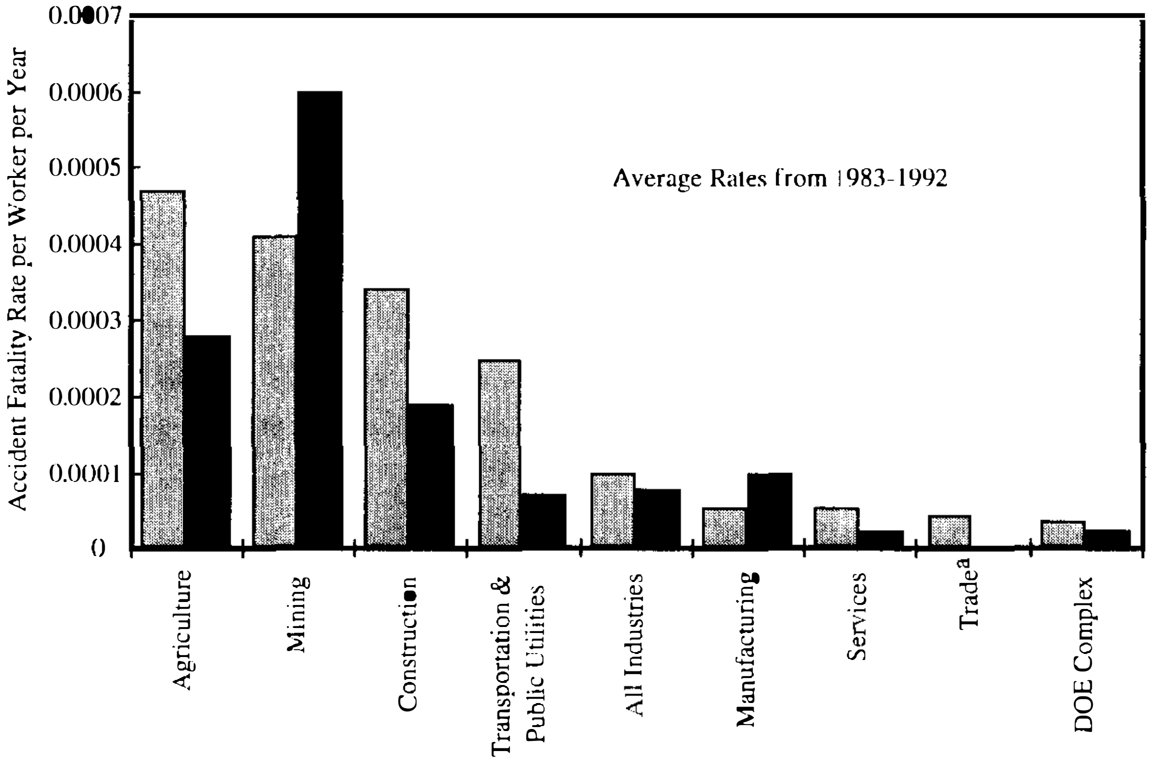
d. The safety analysis report utilized for this accident analysis does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. As demonstrated by the dose to the maximally exposed individual, consequences to the public from this accident could be less than the consequences from Accidents 2 through 4. However, given the high frequency for this accident compared to Accidents 2 through 4, the risk could actually be greater than for Accidents 2 through 4.

e. This attribute is equal to consequences × frequency (events per year). The information is based on conservative (95 percentile) meteorological conditions.

f. ICPP = Idaho Chemical Processing Plant.

g. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

h. Resuming processing at the INEL under this alternative is not considered.



Legend:

U.S.A. Average

State of Idaho

a. Datum for State of Idaho is unavailable.

Sources: NSC (1993); DOE (1993b) and Hendrix (1994)

PJ20-1

Figure 5.15-1. Comparison of fatality rates among workers in various industry groups.

with spent nuclear fuel management activities have occurred. In 1958, filters in the Idaho Chemical Processing Plant CPP-601 Fuel Element Cutting Facility failed during decontamination operations. An estimated 100 curies of particulate radioactivity were released over an area of approximately 200 acres (0.809 square kilometers) in the vicinity of the Idaho Chemical Processing Plant. Approximately 39 curies became airborne, resulting in an estimated dose of 0.11 millirem to a hypothetical offsite individual located at the nearest site boundary (DOE 1991).

Three inadvertent nuclear chain reactions (i.e., nuclear criticalities) occurred at the Idaho Chemical Processing Plant in 1959, 1961, and 1978. The 1959 criticality occurred in a process waste and cell floor drain collection tank. Available evidence indicates that the critical solution resulted from an accidental transfer of concentrated uranyl nitrate solution to the waste collection tank through a line normally used to transfer decontaminating solutions to the waste tank. The estimated airborne release from this incident was 3,700 curies, and the estimated dose to the maximally exposed hypothetical individual located at the nearest site boundary was 1.1 millirem (DOE 1991). The 1961 and 1978 nuclear criticalities resulted from spent nuclear fuel dissolution and reprocessing activities. Estimated releases to the environment as a result of these accidents were 120 curies and 620 curies for the 1961 and 1978 accidents, respectively, and the calculated radiation doses at the nearest site boundary were less than 0.1 millirem for both releases (DOE 1991).

The INEL Fluorinel and Storage (FAST) facility (CPP-666), which historically performed spent nuclear fuel-related reprocessing activities, is currently shut down. Activities are under way to place this facility in a permanent shutdown mode. Restart of this facility and the potential for an inadvertent nuclear criticality resulting from operating this facility are considered in Sections 5.15.4.4 and 5.15.4.5 [Alternatives 4b(1) and 5b, respectively]. Because DOE has no current plans to resume spent nuclear fuel reprocessing activities at the Idaho Chemical Processing Plant, events similar to the three historic nuclear criticalities discussed above will be unlikely in future INEL spent nuclear fuel-related activities. Additional information regarding the historical accidents summarized above is provided in Slaughterbeck et al. (1995).

In the site's 40-year history, three prompt fatalities of INEL workers have occurred by accidents involving radiation exposure. In 1961, a steam explosion resulting from an unplanned nuclear criticality in an experimental reactor (Stationary Low-Power Reactor No. 1) killed these workers, who were manually moving reactor control elements. The estimated dose from this accident to a hypothetical individual located at the nearest site boundary was approximately 3 millirem (DOE 1991). All the accidents discussed above have caused contamination that has led to secondary impacts, such as the contamination of facility equipment and land inside the site boundary, and have required cleanup.

Twenty workers at the Argonne National Laboratory-West facility area were injured in early 1994 when, in an accident involving toxic material exposure, approximately 9 kilograms (20 pounds) of chlorine gas used to treat potable (i.e., drinking) water were accidently released to the environment. Although an investigation into this incident by the DOE was still ongoing at the time this analysis was performed, the accident is presumed to have occurred while a vendor was removing and replacing a nearly empty chlorine cylinder. A maintenance employee assisting in the activity apparently disconnected the nearly empty in-service chlorine gas cylinder from the potable water system with the cylinder valve in the open position, resulting in the remaining tank contents being discharged to the environment. As a result of the accidental release, 20 workers were sent to a local hospital. Eighteen workers reported for treatment of minor respiratory distress, one worker reported symptoms of more serious respiratory problems, and one worker reported back injuries as a result of falling while responding to the accident. (ANL 1994 and DOE 1994b).

5.15.3 Methodology for Determining the Maximum Reasonably Foreseeable Radiological Accidents

5.15.3.1 Selection of Spent Nuclear Fuel Facilities and Operations Requiring Accident Analyses. The accident analyses performed to support this EIS considered all INEL nonreactor nuclear facilities that support spent nuclear fuel-related activities with the exception of those at the Naval Reactors Facility (NRF) area. Appendix D of this EIS discusses each of the spent nuclear fuel management alternatives and postulated accident scenarios associated with the Naval Reactors Facility and other naval spent nuclear fuel facilities.

DOE Order 5480.23 (DOE 1992a) defines nonreactor nuclear facilities as those activities or operations that involve radioactive or fissionable materials in such form and quantity that a nuclear hazard potentially exists to the workers or the general public. This analysis considered spent nuclear fuel facilities designed and constructed as direct support to reactor facilities (e.g., Advanced Test Reactor Storage Canal, which stores spent nuclear fuel and irradiated fuels) as nonreactor spent nuclear fuel facilities.

DOE manages spent nuclear fuel at the following INEL facility areas: Idaho Chemical Processing Plant, Naval Reactors Facility, Test Reactor Area, Auxiliary Reactor Area/Power Burst Facility, Argonne National Laboratory-West, and Test Area North. For further information regarding the activities conducted in these areas, refer to Chapter 2. After identifying all the nonreactor nuclear facilities within these facility areas that stabilize, handle, or store spent nuclear fuel, this analysis ranked the facilities according to potential hazards using preexisting facility "hazard classifications." DOE Order 5480.23 requires contractors operating nonreactor nuclear facilities to perform a hazard classification of a facility to assess the consequences of an unmitigated release of radioactive or hazardous material in one of the following categories[1]:

- Category 1. The hazard analysis shows the potential for significant offsite consequences.
- Category 2. The hazard analysis shows the potential for significant onsite consequences.
- Category 3. The hazard analysis shows the potential for only significant localized consequences.

The classification of nonreactor nuclear facilities in one of these three categories was in accordance with DOE Standard DOE-STD-1027-92 (DOE 1992b). This standard provides guidance for the hazard categorization of nuclear facilities based on facility inventories of radionuclides and the potential for those radionuclides to affect workers or the public if released to the environment.

This analysis used these categories as a screening threshold to identify those facilities of interest (i.e., those spent nuclear fuel-related facilities with sufficient quantities of radionuclides to present the potential for significant impacts to workers or the public if released to the environment). The analysis excluded (screened out) Category 3 (low hazard) facilities if they present possible worker consequences enveloped by postulated accidents at Category 2 facilities. Facilities with a hazard classification of 2 or greater (or Category 3 facilities that were not screened out) were evaluated further, as discussed in the next section.

5.15.3.2 Determination of Maximum Reasonably Foreseeable Radiological Accidents. After determining spent nuclear fuel-related facilities with sufficient quantities of radionuclides to present radiological consequences to workers or the public (as discussed in

1 These categories were formerly labeled "high," "moderate," and "low" in accordance with DOE Order 5481.1B (DOE 1987), which has been superseded by DOE Order 5480.23 for nonreactor nuclear facilities.

Section 5.15.3.1), the analysis generated potential accident scenarios for each of these INEL facilities by performing the following activities:

- Reviewing historic spent nuclear fuel-related accidents that have occurred during the 40-year history of the INEL.
- Reviewing existing accident analyses and safety analysis reports for spent nuclear fuel-related activities and facilities.
- Identifying potential internal, external, and natural phenomena events that could initiate spent nuclear fuel-related accidents other than those previously analyzed.
- Performing additional accident analyses for those accidents considered to present the greatest consequences to workers or the public, as necessary.

The analysis considered internal and external initiators associated with a wide range of activities (e.g., research and development and construction or modification of facilities) not necessarily covered in existing safety analyses. For example, potential radiological accident scenarios initiated by construction activities associated with constructing new spent nuclear fuel-related facilities or modifying existing spent nuclear fuel-related facilities (as proposed under the various alternatives) were postulated. Typically, events involved in the construction of new spent nuclear fuel-related facilities would act as external initiators to existing facilities, while events involved in modifying existing spent nuclear fuel facilities would act as internal initiators. Examples of construction or industrial-type events that could initiate a radiological accident included fires, confinement impacts or puncture events, equipment failure, and human error.

Additional considerations used to determine potential internal and external initiators that could lead to spent nuclear fuel-related radiological accidents included vulnerabilities associated with handling, stabilizing, and storing severely degraded spent nuclear fuel and equipment. For example, in November 1993, DOE issued a report (DOE 1993c) discussing vulnerabilities associated with various spent nuclear fuel-related facilities across the DOE complex. The report identified one INEL facility, the CPP-603 Underwater Fuel Storage Facility, as requiring immediate management attention to avoid unnecessary increases in worker exposures, cleanup costs, and postulated accident frequencies. Activities have begun to stabilize spent nuclear fuel inventories in the CPP-603 facility and relocate them to another facility (CPP-666); these activities will continue for several years after the scheduled

1995 Record of Decision for this EIS. Therefore, the analysis considered postulated accident scenarios associated with stabilizing and relocating CPP-603 spent nuclear fuel inventories to be potential accident initiators in developing the radiological accidents summarized in this EIS. Examples of accident scenarios considered as a result of degraded spent nuclear fuel or facility equipment included inadvertent nuclear criticalities, physical damage of spent nuclear fuel and spent nuclear fuel facilities, and radionuclide releases resulting from handling and stabilizing degraded spent nuclear fuel. For postulated accident scenarios at facilities other than the CPP-603 Underwater Fuel Storage Facility, the analysis also considered the potential for long-term degradation of facility structures, equipment, and spent nuclear fuel inventories that could lead to an increased probability for radiological accidents.

To compare the various possible spent nuclear fuel-related accident scenarios and to identify those maximum reasonably foreseeable accidents that present the greatest consequences to workers and the public, the analysis divided each postulated spent nuclear fuel-related accident into the appropriate frequency category (abnormal events, design-basis accidents[2], or beyond-design-basis accidents), according to its estimated frequency of occurrence. Table 5.15-5 lists the frequency ranges associated with the abnormal event, design-basis accident, and beyond-design-basis accident categories discussed in Section 5.15.1.

The estimated frequency of each postulated accident was based on an identification of the physical basis for the accident and the events required for the accident to occur. Because many of the postulated accidents or their constituent events (initiators or precursors) have rarely or never occurred, frequency data based on historic experience were not available. Therefore, in many instances, it was necessary to develop a frequency estimate on the basis of events for which experience existed and engineering judgment. More than 40 sources of frequency data for the accident events postulated were reviewed, including analyses and reports prepared for the DOE, U.S. Nuclear Regulatory Commission (NRC), Electric Power Research Institute, and private industry. [For further information regarding the development of estimated accident frequencies, refer to Slaughterbeck et al. (1995).]

After the division of the postulated spent nuclear fuel-related accidents into the frequency ranges defined in Table 5.15-5, the analysis identified the postulated nonprocessing-related accident within each frequency range determined to present the maximum offsite consequences as a maximum

2 For facilities where design-basis accident analyses were unavailable, evaluation basis accident scenarios (postulated accident scenarios used where documented design basis accident analyses do not exist) were considered in accordance with DOE-DP-STD-3005-YR (DOE 1994a).

Table 5.15-5. Accident frequency categories.

Frequency Category	Accident Frequency Range (accidents per year)
Abnormal events	frequency $\geq 1\times10^{-3}$ per year
Design-basis accidents	1×10^{-3} per year > frequency $\geq 1\times10^{-6}$ per year
Beyond-design-basis accidents	1×10^{-6} per year > frequency $\geq 1\times10^{-7}$ per year

reasonably foreseeable radiological accident to be further analyzed for this EIS. Potential nonprocessing-related accident scenarios were chosen as maximum reasonably foreseeable accidents because of the shutdown status of the INEL facility (CPP-666) that historically processed spent nuclear fuel. However, because existing inventories of spent nuclear fuel at the INEL would substantially increase under Alternatives 4b(1) and 5b [Regionalization by Geography (INEL) and Centralization at the INEL, respectively], there could be a need to resume processing operations to stabilize degraded spent nuclear fuel operations and assure adequate storage space for spent nuclear fuel received from other sites.[3] Therefore, in addition to the maximum reasonably foreseeable nonprocessing-related accident scenarios, this analysis considers the three postulated processing-related accidents that present the maximum offsite consequences as additional maximum reasonably foreseeable accidents under Alternatives 4b(1) and 5b.

In addition, a postulated inadvertent nuclear criticality accident at the CPP-603 Underwater Storage Facility was considered for further analysis because significant vulnerabilities associated with its spent nuclear fuel inventories have been identified (DOE 1993b) and postulated criticality accidents have been addressed in virtually all nonreactor DOE EISs and safety analysis reports where the accidents are reasonably foreseeable because of public concerns regarding their potential. As a result, the seven radiological accidents summarized in Section 5.15.4 were determined to be the maximum reasonably foreseeable radiological accidents (i.e., greatest consequences). Further discussion and analysis information for each of these accidents, as well as other accidents analyzed, is provided in Slaughterbeck et al. (1995). Appendix D identifies maximum reasonably foreseeable accidents associated with transporting, receiving, handling, and storing naval spent nuclear fuel at the INEL. The postulated accidents summarized in this section considered with the INEL facilities analyzed in

3 Processing would be performed in the Fluorinel and Storage (FAST) facility (CPP-666) and a new facility to be constructed, the Fuel Processing Restoration (FPR) facility (CPP-691). Processing would consist of dissolving spent nuclear fuel to immobilize radionuclides for final waste disposal.

Appendix D provide a basis for characterizing the potential risks and consequences associated with managing spent nuclear fuel at the INEL over the next 40 years.

Seismic events were the only identified common-cause initiators with the potential to initiate radioactive material releases to the environment at more than one spent nuclear fuel-related facility at the INEL. However, a seismic event resulting in significant damage and radioactive releases from facilities in more than one facility area (e.g., Idaho Chemical Processing Plant and Test Area North) is considered beyond reasonably foreseeable (frequency less than one in ten million years), because of the physical distance and isolation between facility areas. In accordance with DOE guidance (DOE 1994a), a seismic event initiating multiple-facility releases in more than one facility area on the site was screened from further consideration because of its extremely low frequency of occurrence.

Analyses were performed that evaluated the potential consequences and risks associated with multiple-facility releases within a single INEL facility area resulting from a severe seismic event (Slaughterbeck et al. 1995). For example, within a 500-meter radius in the Idaho Chemical Processing Plant facility area, there are several spent nuclear fuel facilities, the primary facilities being the CPP-749 dry storage facilities and the CPP-666 and CPP-603 underwater fuel storage facilities. An analysis was performed (Slaughterbeck et al. 1995) to determine whether simultaneous releases from these facilities could result from a severe seismic event. Because the CPP-666 and CPP-749 facilities were designed and qualified to withstand a severe seismic event, they are not expected to contribute to the consequences and risks resulting from a severe seismic event impacting the Idaho Chemical Processing Plant. However, because of known structural deficiencies and vulnerabilities with the spent nuclear fuel at the CPP-603 facility, the CPP-603 facility is expected to be significantly damaged following a severe seismic event, resulting in one or more criticalities and the leakage of contaminated basin water to the surrounding environment. While the consequences from these simultaneous multiple-release mechanisms (one or more criticalities and water drainage) would be greater than the single criticality analyzed for CPP-603 facility (Section 5.15.3.3.2), the consequences and risk of such releases are expected to be bounded by the other accidents analyzed in the EIS--primarily, a seismic event that causes fuel melting at the Argonne National Laboratory-West Hot Fuel Examination Facility (highest consequence accident), and a fuel handling accident in the same facility (highest risk accident, where risk = consequence x frequency). Similar analyses (DOE 1993a) for the Test Area North and Argonne National Laboratory-West also demonstrate that potential multiple-facility releases or multiple-release mechanisms from a single facility resulting from a severe seismic event would also be bounded by accidents postulated for the Hot Fuel Examination Facility. Based on this conclusion and the accident selection methodology described 5.15.3.1, the consequences and risks associated with multiple-facility releases were screened from further consideration since they do not represent the bounding accident scenarios within the frequency categories defined in Table 5.15-5.

In addition, the screening methodology did not specifically include potential accident scenarios associated with operating new spent nuclear fuel handling and storage facilities proposed under the various alternatives considered in this EIS because postulated accident scenarios for existing facilities would bound the consequences associated with potential accidents at new facilities. This assumption is appropriate for two primary reasons. First, the missions of new spent nuclear fuel facilities would be similar to the missions of existing spent nuclear fuel-related DOE facilities, which implies that DOE would consider the same types of accident scenarios for the new facilities it considered for the existing facilities. Second, DOE would design and build new facilities that would incorporate modern preventive and mitigative features to reduce the frequency and potential consequences associated with postulated accidents.

To compare the consequences of the same accident scenario at an identical hypothetical facility constructed at each DOE site included in this EIS (based on local geological and meteorological conditions), Appendix D summarizes postulated accident scenarios for a new Expended Core Facility at Oak Ridge, Hanford Site, Savannah River Site, or Nevada Test Site.

To determine the radiological and toxicological consequences presented throughout Section 5.15 associated with the postulated accidents and with spent nuclear fuel-related activities, the analysis used the following definitions:

- Worker. An individual 100 meters (328 feet) downwind of the facility location where the release occurs.[4]

- Nearest Public Access. The nearest point of public access to the location where the release occurs, sometimes inside the site boundary.

4 The worker is defined as the individual located at 100 meters because reliable safety analyses quantifying the impacts (e.g., dose and health effects) to workers at distances less than 100 (i.e., "close-in" workers) meters from an accidental release of radionuclides are unavailable. The effects on and risks to workers closer in than 100 meters are recognized and discussed in Section 5.15.3.3. Each of the maximum reasonably foreseeable accidents considered in this EIS, particularly the design-basis and beyond-design-basis accidents, contains some risk of worker injury or death at distances closer than 100 meters.

- Maximally Exposed Offsite Individual. A hypothetical resident at the site boundary nearest to the facility where the release occurs.

- Offsite Population. The collective total of individuals within an 80-kilometer (50-mile) radius of the INEL.

- Environment. The area outward from 100 meters (328 feet) downwind of the facility where the release occurs.

5.15.3.3 Impact of Accidents on Close-In Workers. An evaluation has been made on the radiological impact to close-in workers from the selected accident scenarios. Injuries or fatalities that might occur due to an external event, such as a severe seismic disturbance or airplane crash into the structure, are not considered in this evaluation since they are not attributable to direct radiological consequences. Seven accident scenarios for nonprocessing-related and processing-related activities are considered maximum reasonably foreseeable accidents.

5.15.3.3.1 Mechanical Handling Accident at the Argonne National Laboratory West Hot Fuel Examination Facility — This accident is assumed to result in fuel pin breach and venting of noble gases and iodine. No fatalities to workers are expected from this event. However, a substantial iodine dose to the thyroid could cause radiation-induced hypothyroidism or a similar disorder.

5.15.3.3.2 Criticality Accident at the Idaho Chemical Processing Plant - CPP-603 — This event is an unplanned nuclear criticality associated with underwater spent nuclear fuel storage at the CPP-603 facility. Based on shielding provided by the pool water, it is likely that no fatalities would occur. To the extent water is expelled due to the energy of the event, close-in workers could receive substantial radiation exposure. Worker presence in the area above the pool or very close to the edge of the pool is not routine. The impact of the event would likely be isolated to nearby equipment operators if the criticality were initiated by a handling error.

5.15.3.3.3 Seismic Event Leading to Fuel Melt at the Argonne National Laboratory West Hot Fuel Examination Facility — A seismic event is postulated to result in a breech of the main cell used for examination of the fuel, which is assumed to lead to a failure of the fuel cooling system. It is likely that the release of radioactive materials from fuel melting would occur slowly enough to allow evacuation of all workers before any appreciable exposure. Therefore, no radiation-induced fatalities would be expected.

5.15.3.3.4 Airplane Crash and Fire at Argonne National Laboratory West Hot Fuel Examination Facility — An airplane crash and subsequent fire sustained by airplane fuel could result in a major breach of the confinement barriers and could lead to a substantial atmospheric release of radionuclides. Workers unaffected by the airplane crash or fire would not be expected to remain in the area long enough to receive substantial radiation exposure. It is assumed the buoyancy of the radioactive material due to the fire would mitigate the direct radiological impacts to close-in workers, substantially reducing the likelihood of radiation induced worker fatalities.

5.15.3.3.5 Criticality Accident During Processing at the Idaho Chemical Processing Plant - CPP-666 — This is the first of three evaluated accidents that could occur only if processing were resumed at the Fluorinel and Storage Facility (FAST). Three inadvertent nuclear criticalities have occurred in INEL processing facilities and none has resulted in worker fatalities. In each event, radioactive material was released to the atmosphere and close-in workers received direct exposure. If processing were resumed, the techniques and controls implemented to prevent recurrence of processing-related criticalities would be employed again. Due to the cell wall shielding provided by concrete walls that are several feet thick, it is expected that no workers would receive substantial radiation exposure.

5.15.3.3.6 Hydrogen Explosion at the Idaho Chemical Processing Plant — A hydrogen explosion in the dissolver off-gas system of the Flourinel and Storage (FAST) Facility would result in release of radioactive material to the facility. If workers were near the dissolver off-gas system, they could receive substantial radiation exposure from the explosion. No fatalities would be expected, but radiation-induced health detriments could occur.

5.15.3.3.7 Dissolution of Short-Cooled Fuel at the Idaho Chemical Processing Plant — An explosion in the dissolver tank could occur if fuel that has not cooled for at least 30 days was inadvertently shipped to the dissolver at the Flourinel and Storage Facility (FAST). This energetic event would likely breach the dissolver off gas system and could breach the dissolver tank. Workers in the areas closely associated with the dissolver tank could receive substantial radiation exposure, but it is likely that no radiation-induced fatalities would occur.

5.15.3.4 Analysis of Radiological Accident Consequences. The quantities of radioactive materials and the ways these materials interact with human beings are important factors in determining health effects. The ways in which radioactive materials reach human beings, their absorption and retention in the body, and the resulting health effects have been studied in great detail. The International Commission on Radiological Protection (ICRP) has made specific recommendations for quantifying these health effects (ICRP 1991). This organization is the recognized body for establishing standards for the protection of workers and the public from the effects of radiation exposure. Health effects can be classified into two categories: prompt (also referred to as acute) and latent. Prompt health effects are those experienced immediately after exposure and include damage to the body up to and including death. Latent health effects are those experienced some time after exposure and include cancers and hereditary symptoms. An INEL-developed computer code, Radiological Safety Analysis Computer Program-5 (RSAC-5), estimates potential radiation doses to maximally exposed individuals or population groups from accidental releases of radionuclides. This code, which is customized to specific INEL conditions, uses well-established and generally accepted scientific engineering principles as the basis for its various calculational steps. The code is based on guidance provided in NRC Guide 1.145 (NRC 1983) and has been validated to comply with accepted standards for such software. [For a detailed description of RSAC-5, refer to Slaughterbeck et al. (1995).]

The RSAC-5 code determined estimated consequences to the worker, an individual assumed to be stranded at the nearest point of public access, the maximally exposed hypothetical individual at the nearest site boundary, and the offsite population within 80 kilometers (50 miles) of the radiological accidents postulated under Alternative 1, No Action. Postulated frequencies and consequences analyzed under Alternative 1 are based on (1) the approximate amount of spent nuclear fuel currently at the INEL [measured in Metric Tons Heavy Metal (MTHM)], (2) the estimated increases in inventories resulting from spent nuclear fuel generated by operating INEL reactors (i.e., fuel recently removed from a reactor that has not had sufficient time to cool), and (3) the estimated number of fuel handling activities associated with stabilizing or relocating spent fuel inventories inside the INEL site boundary. Although the four nonprocessing-related maximum reasonably foreseeable radiological accident scenarios identified for Alternative 1 are also considered under Alternatives 2 through 5, proposed changes in INEL spent nuclear fuel inventories and the number of fuel handling activities associated with these changes could affect the estimated frequencies and consequences expected for Alternatives 2 through 5. Therefore, to reasonably estimate the frequencies and consequences associated with activities proposed under Alternatives 2 through 5, the frequencies and consequences for the accidents presented under Alternative 1 require appropriate "adjustment" or "scaling."

To be conservative, the analysis assumed that the increase in the annual frequency of mechanical handling accidents would be equal to the estimated increase in the annual number of handling events proposed under Alternatives 2 through 5. However, the consequences associated with a mechanical handling accident would not vary with a change in the number of handling events because the amount of material involved in each event would not change. To determine potential changes in annual mechanical handling accident frequencies between the different spent nuclear fuel management alternatives, the analysis based its estimates of the annual number of fuel handling events under each alternative on spent fuel shipment rates anticipated for the next 40 years, as discussed in Appendix I. Estimates of long-term (40-year) and short-term (5-year) shipments at the INEL were considered in determining the annual shipment rates for each alternative. The basis for the number of long-term shipments include spent nuclear fuel the INEL will continue to receive from operating reactors such as DOE, Naval Nuclear Propulsion Program, university, and research reactors. Short-term shipments consist of shipments that would be required to relocate existing spent fuel inventories between sites under the various alternatives. Table 5.15-6 summarizes the estimated annual shipment rate to and from the INEL under each alternative, and within INEL site boundaries. The estimates provided in Table 5.15-6 consider both onsite and offsite shipments.

Table 5.15-6. Determination of accident frequency adjustment factors for Alternatives 2 through 5 based on estimated number of annual spent nuclear fuel shipments under each alternative.[a]

	Alternative	Estimated Shipment Rate (per year)[a]	Adjustment Factor (shipment rate/baseline)
1.	No Action	41	Baseline
2.	Decentralization	50	1.2
3.	1992/1993 Planning Basis	128	3.1
4a.	Regionalization by Fuel Type	195	4.8
4b(1)	Regionalization by Geography (INEL)	824	20.0
4b(2)	Regionalization by Geography (Elsewhere)	351	8.6
5a.	Centralization at Other DOE Sites	351	8.6
5b.	Centralization at the INEL	824	20.0

a. Data presented for the estimated annual shipment rate is based on information tabulated in Appendix I. The annual shipment rate for the No-Action Alternative (baseline) is derived from Table 3 of Wichmann 1994.

Based on the number of annual shipments estimated for Alternatives 2 through 5, as listed in Table 5.15-6, the analysis calculated multiplication factors by dividing the estimated shipment rates under Alternatives 2 through 5 by the baseline (Alternative 1) shipment rate. To determine the estimated frequency for the maximum reasonably foreseeable mechanical handling accidents under each alternative, the frequency identified for Alternative 1 was multiplied by the appropriate adjustment factor. The same approach determined estimated frequencies for Accident 1 (fuel pin breach and noble gases and iodine release from the Hot Fuel Examination Facility) under Alternatives 2 through 5. For Accident 2 (inadvertent criticality in the CPP-603 Underwater Fuel Storage Facility resulting from a handling accident associated with degraded spent nuclear fuel), the estimated frequency considered under Alternative 1 (1×10^{-3} event per year) is based on the number of handling activities associated with relocation of the CPP-603 spent nuclear fuel inventories to the CPP-666 facility. Because proposed changes in INEL inventories under the different alternatives would not affect handling events associated with relocating spent fuel from the CPP-603 facility to the CPP-666 facility, the estimated frequency for this mechanical handling event would not change. As a result of this approach and the fact that 3 of the 4 accident scenarios that present the greatest consequences are not handling accidents, Accident 1 is the only accident requiring "adjustment" for each alternative.

Variable source-term-sensitive accidents would have consequences that depended on the amount of spent nuclear fuel in storage. One example is the accidental drainage of a spent fuel storage canal that results in the release of corrosion products in the canal to the environment. The larger the spent fuel inventory in the canal, the larger the release of corrosion products to the environment resulting from draining the canal. (Drainage of a water canal completely filled with spent nuclear fuel was considered in the determination of the maximum reasonably foreseeable accidents and was determined to present lower consequences than other accident scenarios analyzed.) Variable source-term sensitive accidents depend only on spent nuclear fuel inventories and do not require adjustment of their estimated frequencies of occurrence. Because none of the postulated accidents summarized under Alternative 1 is source-term sensitive (e.g., spent nuclear fuel inventories in the Hot Fuel Examination Facility are not likely to increase), adjustment of the estimated consequences calculated under Alternative 1 is not required for Alternatives 2 through 5.

5.15.4 Impacts from Postulated Maximum Reasonably Foreseeable Radiological Accidents

Section 5.15.4.1 summarizes impacts (e.g., exposures and health effects) from the four nonprocessing-related maximum reasonably foreseeable radiological accidents postulated under

Alternative 1 (No Action). Sections 5.15.4.4.2.1 through 5.15.4.5.2 describe changes in these postulated accident impacts resulting from changes in spent nuclear fuel inventories and handling activities under the other alternatives. Sections 5.15.4.4.2.1 and 5.15.4.5.2 also summarize impacts from three additional maximum reasonably foreseeable accidents associated with resumption of processing activities at the INEL. Section 5.15.6 provides more information about the assumptions and analyses performed for each of the radiological accidents discussed under each alternative.

5.15.4.1 Alternative 1: No Action. Based on the quantity of spent nuclear fuel at the INEL (excluding naval fuel at Naval Reactors Facility, which is analyzed in Appendix D), its storage configuration (wet versus dry), the amount of time the spent fuel has been allowed to cool, and consideration of various internal, external, and natural phenomena initiators (as discussed in Section 5.15.3), the postulated accidents listed in Table 5.15-7 would have the greatest radiological consequences within the abnormal event, design-basis accident, and beyond-design-accident categories under this alternative. For each accident, Table 5.15-7 also lists estimated accident frequencies; radiation exposures to the offsite population within 80 kilometers (50 miles), a member of the public stranded at the nearest point of public access inside the INEL site boundary, a hypothetical maximally exposed individual (MEI) at the nearest site boundary, and a worker; point estimates of the annualized risk of the maximally exposed individual contracting a fatal cancer during his/her lifetime as a result of the radiation exposure; and point estimates of risk of the expected number of fatal cancers (annualized and total) in the offsite population. The estimates of the consequences and risk to the offsite population are based on conservative (95 percentile) and average (50 percentile) meteorological conditions[5]. The estimates of the consequences and risk to the maximally exposed individual are based on conservative (95 percentile) meteorological conditions. The postulated accidents listed in Table 5.15-7, in conjunction with the maximum reasonably foreseeable spent nuclear fuel accidents identified for the INEL Naval Reactors Facility in Appendix D, characterize the potential consequences and risks associated with the proposed spent fuel management activities at the INEL under this alternative.

Atmospheric transport of radionuclides from the postulated accidents could result in some secondary impacts, such as contamination of the environment or impacts to national defense. To

5 Conservative (95 percentile) meteorological conditions are defined as the meteorological conditions that, for a given release, the concentration at a fixed receptor location will not be exceeded 95 percent of the time. Average (50 percentile) meteorological conditions are defined as the meteorological conditions that, for a given release, the concentration at a fixed receptor location will not be exceeded 50 percent of the time.

Table 5.15-7. Impacts from selected maximum reasonably foreseeable radiological accidents - Alternative 1, No Action (50 and 95 percentile meteorological conditions).

Accident	Frequency (events per year)	Worker Dose[a] (rem)	Nearest Public Access[b] (rem)	Dose to MEI[c] (rem)	Offsite Population Dose (95%) (person-rem)	Point estimates of risk of fatal cancers (per year): MEI 95%[d]	Point estimates of risk of fatal cancers (per year): Offsite Population 50%	Point estimates of risk of fatal cancers (per year): Offsite Population 95%
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[e]	1.0×10^{-2}	(f)	(f)	2.0×10^{-3}	(f)	1.0×10^{-8}	(f)	(f)
2. Inadvertent criticality in ICPP[g] CPP-603 storage facility[h]	1.0×10^{-3}	9.7×10^{-2}	1.4×10^{-3}	1.0×10^{-3}	5.9×10^{-1}	5.0×10^{-10}	6.5×10^{-9} (6.5×10^{-6})[d]	3.0×10^{-7} (3.0×10^{-4})[d]
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	1.0×10^{-5}	6.2×10^{-1}	6.5×10^{-1}	5.0×10^{0}	1.4×10^{4}	2.5×10^{-8}	4.5×10^{-7} (4.5×10^{-2})[d]	7.0×10^{-5} (7.0×10^{0})[d]
4. Material release from HFEF resulting from aircraft crash and ensuing fire	1.0×10^{-7}[(i)]	4.6×10^{0}	3.2×10^{-1}	5.0×10^{0}	2.0×10^{3}	2.5×10^{-10}	3.6×10^{-8} (3.6×10^{-1})[d]	1.0×10^{-7} (1.0×10^{0})[d]

a. A worker is defined as a worker located 100 meters (328 feet) from the point of release.

b. Public individual assumed to be stranded at the nearest point of public access inside the site boundary.

c. MEI = Maximally exposed hypothetical offsite individual, located at the nearest site boundary.

d. Maximally exposed individual and offsite population fatal cancer risk = dose × accident frequency × 5.0×10^{-4} fatal cancer per rem (ICRP-60 conversion factor) if dose is less than 20 rem. For doses 20 rem or more the ICRP-60 conversion factor is doubled, or 1.0×10^{-3}. Numbers in parentheses indicate the total number of fatal cancers in the population if the accident occurred.

e. HFEF - Hot Fuel Examination Facility.

f. The safety analysis report utilized for this accident analysis does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. As demonstrated by the dose to the maximally exposed individual, consequences to the public from this accident could be less than the consequences from Accidents 2 through 4.

g. ICPP = Idaho Chemical Processing Plant.

h. Although three nuclear criticalities associated with spent nuclear fuel reprocessing activities have occurred at the INEL during its 40-year operating history, the estimated frequency for an inadvertent criticality is not based on historic reprocessing data because reprocessing is not considered under this alternative. Nominal frequency estimates vary from 1.0×10^{-4} (CPP-666 underwater storage facility) to 1.0×10^{-3} (CPP-603 underwater storage facility) event per year.

i. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

prevent these radionuclides from increasing any potential safety concerns, DOE would initiate cleanup activities if an accident occurred, and no irreversible environmental impacts would be likely.

Table 5.15-8 summarizes postulated secondary impacts resulting from the postulated radiological accidents listed in Table 5.15-7.

This analysis takes limited credit for emergency response actions in determining the consequences listed in Table 5.15-7. DOE would initiate INEL emergency response programs, as appropriate, following the occurrence of an accident to prevent or mitigate potential consequences. These emergency response programs, implemented in accordance with 5500-DOE series Orders, typically involve emergency planning, emergency preparedness, and emergency response actions. Each emergency response plan utilizes resources specifically dedicated to assist a facility in emergency management. These resources include but are not limited to the following:

- INEL Warning Communications Center
- INEL Fire Department
- Facility Emergency Command Centers
- DOE Emergency Operations Centers
- County and State Emergency Command Centers
- Medical, health physics, and industrial hygiene specialists
- Protective clothing and equipment (respirators, breathing air supplies, etc.)
- Periodic training exercises and drills within and between the organizations involved in implementing the response plans

5.15.4.2 Alternative 2: Decentralization. Adjustments in estimated accident frequencies and point estimates of risk presented for Alternative 1 would be related to (1) the receipt, handling, and storage activities associated with the additional spent nuclear fuel inventories; and (2) the increase in overall spent nuclear fuel-related storage, relocation, and handling activities not allowed under Alternative 1. Because no changes in the accident consequences estimated for Alternative 1 are likely to occur under this alternative from increased fuel inventories (i.e., the same amount of radioactive material would accidentally be released to the environment as discussed in Section 5.15.3.3), no changes are likely in the postulated secondary impacts listed in Table 5-15-8. Table 5.15-9 summarizes the four postulated accidents with the greatest radiological impacts under this alternative.

Table 5.15-8. Estimated secondary impacts resulting from the maximum reasonably foreseeable accidents postulated under Alternative 1, No Action, assuming conservative (95 percentile) meteorological conditions.

Radiological Accident Summary	Environmental or Social Impacts (Assuming 88 millirem per year limit with 24-hour-per-day exposure)[a]							
	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights & Tribal Resources
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[b] (1×10^{-2} per year)	Limited adverse effects expected to vegetation or wildlife.	Limited adverse effects expected to surface water or groundwater.	Limited economic impacts expected. Any cleanup required would be localized and could be accomplished with existing workforce and equipment.	No effects on national defense expected.	Local contamination requiring cleanup expected around site accident.	No impacts exptected to endangered or threatened species.	No change in land use or irreversible impacts expected.	No irreversible impacts to Native Americans or public lands expected.
2. Uncontrolled chain reaction (criticality) at ICPP[c] (1×10^{-3} per year)	Limited adverse effects expected to vegetation or wildlife.	Limited adverse effects expected to surface water or groundwater.	No economic impacts expected. Any cleanup required would be localized and could be accomplished with existing workforce and equipment.	No effects on national defense expected.	Local contamination requiring cleanup expected around site accident.	No impacts exptected to endangered or threatened species.	No change in land use or irreversible impacts expected.	No irreversible impacts to Native American or public lands expected.
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach (1×10^{-5} per year)	Limited adverse effects expected to vegetation or wildlife.	Limited adverse effects expected to surface water or groundwater.	Potential interdiction of affected agricultural products on nearby lands. Local cleanup in the vicinity of HFEF.	No effects on national defense expected.	Local contamination requiring cleanup expected around site accident.	No impacts exptected to endangered or threatened species.	Potential for 1 year of agricultural land withdrawal of up to 10,000 acres[d] (on and off the INEL site).	Potential for temporary restricted access to affected public land (less than 10,000 acres).[d]
4. Material release from HFEF resulting from aircraft crash and ensuing fire (1×10^{-7} per year)	Limited adverse effects expected to vegetation or wildlife.	Limited adverse effects expected to surface water or groundwater.	Potential interdiction of affected agricultural products on nearby lands. Local cleanup in the vicinity of HFEF.	No effects on national defense expected.	Local contamination requiring cleanup expected around site accident.	No impacts exptected to endangered or threatened species.	Potential for 1 year of agricultural withdrawal of up to 10,000 acres[d] (on and off the INEL site).	Potential for temporary restricted access to affected public land (less than 10,000 acres).[d]

a. Postulated secondary unpacts based on 10-microrem-per-hour exposure (88 millirem per year with 24-hour-per-day exposure) from ground contamination resulting from radionuclide deposition from the plume. This approach in estimated secondary impacts is conservative because DOE Order 5400.5 states that the public dose limit for exposure to residual contamination and natural background radiation is 100 millirem per year.

b. HFEF = Hot Fuel Examination Facility.

c. ICPP = Idaho Chemical Processing Plant.

d. To convert acres to square kilometers, multiply by 0.004.

Table 5.15-9. Impacts from selected maximum reasonably foreseeable accidents - Alternative 2, Decentralization (50 and 95 percentile meteorological conditions).

Accident	Adjusted Frequency[a] (events per year)	Worker Dose[b] (rem)	Nearest Public Access[c] (rem)	Dose to MEI[d] (rem)	Offsite Population Dose (95%) (person-rem)	Adjusted point estimates of risk of fatal cancers (per year)		
						MEI	Offsite	Population
						95%[e]	50%	95%
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[f]	1.2×10^{-2} (1.2)	(g)	(g)	2.0×10^{-3}	(g)	1.2×10^{-8}	(g)	(g)
2. Inadvertent criticality in ICPP[h] CPP-603 storage facility[i]	1.0×10^{-3} (1.0)[j]	9.7×10^{-2}	1.4×10^{-3}	1.0×10^{-3}	5.9×10^{-1}	5.0×10^{-10}	6.5×10^{-9} $(6.5\times10^{-6})^e$	3.0×10^{-7} $(3.0\times10^{-4})^e$
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	1.0×10^{-5} (1.0)	6.2×10^{-1}	6.5×10^{-1}	5.0×10^{0}	1.4×10^{4}	2.5×10^{-8}	4.5×10^{-7} $(4.5\times10^{-2})^e$	7.0×10^{-5} $(7.0\times10^{0})^e$
4. Material release from HFEF resulting from aircraft crash and ensuing fire	$1.0\times10^{-7(k)}$ (1.0)	4.6×10^{0}	3.2×10^{-1}	5.0×10^{0}	2.0×10^{3}	2.5×10^{-10}	3.6×10^{-8} $(3.6\times10^{-1})^e$	1.0×10^{-7} $(1.0\times10^{0})^e$

a. Numbers in parentheses indicate multiplication factor used to scale or adjust estimated accident frequencies under Alternative 1, as described in Section 5.15.3.3.
b. A worker is defined as a worker located 100 meters (328 feet) from the point of release.
c. Public individual assumed to be stranded at the nearest point of public access inside the site boundary.
d. MEI = Maximally exposed hypothetical offsite individual located at the nearest site boundary.
e. Maximally exposed individual and offsite population fatal cancer risk = dose × accident frequency × 5.0×10^{-4} fatal cancer per rem (ICRP-60 conversion factor) if dose is less than 20 rem. For doses of 20 rem or more, the ICRP-60 conversion factor is doubled, or 1.0×10^{-3}. Numbers in parentheses indicate total number of fatal cancers in the population if the accident occurs.
f. HFEF = Hot Fuel Examination Facility.
g. The safety analysis report utilized for this accident analysis does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. As demonstrated by the dose to the maximally exposed individual, consequences to the public from this accident could be less than the consequences from Accidents 2 through 4.
h. ICPP = Idaho Chemical Processing Plant.
i. Although three nuclear criticalities associated with spent nuclear fuel reprocessing activities have occurred at the INEL during its 40-year operating history, the estimated frequency for an inadvertent criticality is not based on historic reprocessing data since reprocessing is not considered under this alternative. Nominal frequency estimates vary from 1.0×10^{-4} (CPP-666 underwater storage facility) to 1.0×10^{-3} (CPP-603 underwater storage facility) events per year.
j. Refer to Sections 5.15.3.3 and 5.15.6.2 for details on why this frequency was not adjusted under this alternative.
k. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

5.15.4.3 Alternative 3: 1992/1993 Planning Basis. Under this alternative, the INEL could receive the following spent nuclear fuel:

- Spent nuclear fuel from domestic DOE and university reactors and foreign research test reactors

- All Training Reactor Isotopics General Atomics (TRIGA) spent nuclear fuel from foreign and Hanford reactors

- Fort St. Vrain spent nuclear fuel from Public Service Company of Colorado

- Special case commercial pressurized water reactor and boiling water reactor spent nuclear fuel from West Valley, New York

- Naval spent nuclear fuel from sites such as the Norfolk or Puget Sound Naval Shipyard.

Adjustments in estimated accident frequencies and point estimates of risk presented for Alternative 1 would be related to (1) the receipt, handling, and storage activities associated with the additional spent nuclear fuel inventories; and (2) the increase in overall spent fuel-related storage, relocation, and handling activities not allowed under Alternative 1. Because no changes in the accident consequences estimated for Alternative 1 are likely to occur under this alternative from increased fuel inventories (i.e., the same amount of radioactive material would accidentally be released to the environment as discussed in Section 5.15.3.3), no changes are likely in the postulated secondary impacts listed in Table 5.15-8. Table 5.15-10 summarizes the postulated accidents with the greatest radiological impacts under this alternative.

5.15.4.4 Alternative 4: Regionalization. Under this alternative, there are two primary Regionalization alternatives: (1) Alternative 4a (Regionalization by Fuel Type), where existing and spent nuclear fuel inventories will be distributed between the DOE sites based primarily on the similarity of fuel types, although DOE would also consider transportation distances, available stabilization capabilities, available storage capacities, or a combination of these factors; or (2) Alternative 4b (Regionalization by Geography), where existing and new spent nuclear fuel inventories in the western region of the country will be centralized at a single western site, and existing and new spent nuclear fuel inventories in the eastern region of the country will be centralized at a single eastern site.

Table 5.15-10. Impacts from selected maximum reasonably foreseeable accidents - Alternative 3, Planning Basis (50 and 95 percentile meteorological conditions).

Accident	Adjusted Frequency[a] (events per year)	Worker Dose[b] (rem)	Nearest Public Access[c] (rem)	Dose to MEI[d] (rem)	Offsite Population Dose (95%) (person-rem)	Adjusted point estimates of risk of fatal cancers (per year)		
						MEI	Offsite Population	
						95%[e]	50%	95%
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[f]	3.1×10^{-2} (3.1)	(g)	(g)	2.0×10^{-3}	(g)	3.1×10^{-8}	(g)	(g)
2. Inadvertent criticality in ICPP[h] CPP-603 storage facility[i]	1.0×10^{-3} (1.0)[j]	9.7×10^{-2}	1.4×10^{-3}	1.0×10^{-3}	5.9×10^{-1}	5.0×10^{-10}	6.5×10^{-9} $(6.5\times10^{-6})^e$	3.0×10^{-7} $(3.0\times10^{-4})^e$
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	1.0×10^{-5} (1.0)	6.2×10^{-1}	6.5×10^{-1}	5.0×10^{0}	1.4×10^{4}	2.5×10^{-8}	4.5×10^{-7} $(4.5\times10^{-2})^e$	7.0×10^{-5} $(7.0\times10^{0})^e$
4. Material release from HFEF resulting from aircraft crash and ensuing fire	$1.0\times10^{-7(k)}$ (1.0)	4.6×10^{0}	3.2×10^{-1}	5.0×10^{0}	2.0×10^{3}	2.5×10^{-10}	3.6×10^{-8} $(3.6\times10^{-1})^e$	1.0×10^{-7} $(1.0\times10^{0})^e$

a. Numbers in parentheses indicate multiplication factor used to scale or adjust estimated accident frequencies under Alternative 1, as described in Section 5.15.3.3.
b. A worker is defined as a worker located 100 meters (328 feet) from the point of release.
c. Public individual assumed to be stranded at the nearest point of public access inside the site boundary.
d. MEI = Maximally exposed hypothetical offsite individual located at the nearest site boundary.
e. Maximally exposed individual and offsite population fatal cancer risk = dose × accident frequency × 5.0×10^{-4} fatal cancer per rem (ICRP-60 conversion factor) if dose is less than 20 rem. For doses of 20 rem or more, the ICRP-60 conversion factor is doubled, or 1.0×10^{-3}. Numbers in parentheses indicate total number of fatal cancers in the population if the accident occurs.
f. HFEF = Hot Fuel Examination Facility.
g. The safety analysis report utilized for this accident analysis does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. As demonstrated by the dose to the maximally exposed individual, consequences to the public from this accident could be less than the consequences from Accidents 2 through 4. However, given the high frequency for this accident compared to Accidents 2 through 4, the risk could actually be greater than for Accidents 2 through 4.
h. ICPP = Idaho Chemical Processing Plant.
i. Although three nuclear criticalities associated with spent nuclear fuel reprocessing activities have occurred at the INEL during its 40-year operating history, the estimated frequency for an inadvertent criticality is not based on historic reprocessing data since reprocessing is not considered under this alternative. Nominal frequency estimates vary from 1.0×10^{-4} (CPP-666 underwater storage facility) to 1.0×10^{-3} (CPP-603 underwater storage facility) events per year.
j. Refer to Sections 5.15.3.3 and 5.15.6.2 for details on why this frequency was not adjusted under this alternative.
k. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

5.15.4.4.1 Alternative 4a - Regionalization By Fuel Type — Adjustments in the estimated accident frequencies and point estimates of risk presented for Alternative 1 would be related to (1) the receipt, handling, and storage activities associated with the additional spent nuclear fuel inventories; and (2) the increase in overall spent nuclear fuel-related storage, relocation, and handling activities not allowed under Alternative 1. Because no changes in the accident consequences estimated for Alternative 1 are likely to occur under this alternative from increased fuel inventories (i.e., the same amount of radioactive material would accidentally be released to the environment as discussed in Section 5.15.3.3), no changes are likely in the postulated secondary impacts listed in Table 5.15-8. Table 5.15-11 summarizes the postulated accidents with the greatest radiological impacts under this alternative.

5.15.4.4.2 Alternative 4b - Regionalization by Geography — Under this alternative, spent nuclear fuel inventories in the western region of the country would be centralized at either the INEL, Hanford Site, or Nevada Test Site. Alternative 4b(1) considers regionalization at the INEL. Alternative 4b(2) considers regionalization at the Hanford Site or Nevada Test Site.

5.15.4.4.2.1 Alternative 4b(1) - Regionalization by Geography (INEL) — Under this alternative, existing and new spent nuclear fuel inventories in the western region of the country would be centralized at the INEL. Fuel stabilization would be performed in the Fluorinel and Storage (FAST) facility (CPP-666) and a new facility to be constructed, the Fuel Processing Restoration facility (CPP-691), to dissolve spent nuclear fuel and stabilize (i.e., immobilize) radionuclides. Because the volume of spent nuclear fuel considered under this alternative is only slightly lower than that considered under Alternative 5b, adjustments in the estimated accident frequencies and point estimates of risk for the four accidents presented under Alternative 1 were conservatively considered equivalent to the adjustments required under Alternative 5b (i.e., centralization of all the DOE, Naval Nuclear Propulsion Program, university, and research reactor spent nuclear fuel in the country at the INEL). Adjustments in the estimated accident frequencies and point estimates of risk for the four accidents presented under Alternative 1 would be related to (1) the receipt, handling, and storage activities associated with the additional spent nuclear fuel inventories; and (2) the increase in overall spent nuclear fuel-related storage, relocation, and handling activities not allowed under Alternative 1. Because no changes in the accident consequences estimated for Alternative 1 are likely to occur under this alternative from increased fuel inventories (i.e., the same amount of radioactive material would accidentally be released to the environment as discussed in Section 5.15.3.3), no changes are likely in the postulated secondary impacts listed in Table 5.15-8.

Table 5.15-11. Impacts from selected maximum reasonably foreseeable accidents - Alternative 4a, Regionalization by Fuel Type (50 and 95 percentile meteorological conditions).

Accident	Adjusted Frequency[a] (events per year)	Worker Dose[b] (rem)	Nearest Public Access[c] (rem)	Dose to MEI[d] (rem)	Offsite Population Dose (95%) (person-rem)	Adjusted point estimates of risk of fatal cancers (per year): MEI 95%[e]	Offsite Population 50%	Offsite Population 95%
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[f]	4.8×10^{-2} (4.8)	(g)	(g)	2.0×10^{-3}	(g)	4.8×10^{-8}	(g)	(g)
2. Inadvertent criticality in ICPP[h] CPP-603 storage facility[i]	1.0×10^{-3} (1.0)[j]	9.7×10^{-2}	1.4×10^{-3}	1.0×10^{-3}	5.9×10^{-1}	5.0×10^{-10}	6.5×10^{-9} (6.5×10^{-6})[e]	3.0×10^{-7} (3.0×10^{-4})[e]
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	1.0×10^{-5} (1.0)	6.2×10^{-1}	6.5×10^{-1}	5.0×10^{0}	1.4×10^{4}	2.5×10^{-8}	4.5×10^{-7} (4.5×10^{-2})[e]	7.0×10^{-5} (7.0×10^{0})[e]
4. Material release from HFEF resulting from aircraft crash and ensuing fire	1.0×10^{-7}[k] (1.0)	4.6×10^{0}	3.2×10^{-1}	5.0×10^{0}	2.0×10^{3}	2.5×10^{-10}	3.6×10^{-8} (3.6×10^{-1})[e]	1.0×10^{-7} (1.0×10^{0})[e]

a. Numbers in parentheses indicate multiplication factor used to scale or adjust estimated accident frequencies under Alternative 1, as described in Section 5.15.3.3.

b. A worker is defined as a worker located 100 meters (328 feet) from the point of release.

c. Public individual assumed to be stranded at the nearest point of public access inside the site boundary.

d. MEI = Maximally exposed hypothetical offsite individual located at the nearest site boundary.

e. Maximally exposed individual and offsite population fatal cancer risk = dose × accident frequency × 5.0×10^{-4} fatal cancer per rem (ICRP-60 conversion factor) if dose is less than 20 rem. For doses of 20 rem or more, the ICRP-60 conversion factor is doubled, or 1.0×10^{-3}. Numbers in parentheses indicate total number of fatal cancers in the population if the accident occurs.

f. HFEF = Hot Fuel Examination Facility.

g. The safety analysis report utilized for this accident analysis does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. As demonstrated by the dose to the maximally exposed individual, consequences to the public from this accident could be less than the consequences from Accidents 2 through 4. However, given the high frequency for this accident compared to Accidents 2 through 4, the risk could actually be greater than for Accidents 2 through 4.

h. ICPP = Idaho Chemical Processing Plant.

i. Although three nuclear criticalities associated with spent nuclear fuel reprocessing activities have occurred at the INEL during its 40-year operating history, the estimated frequency for an inadvertent criticality is not based on historic reprocessing data since reprocessing is not considered under this alternative. Nominal frequency estimates vary from 1.0×10^{-4} (CPP-666 underwater storage facility) to 1.0×10^{-3} (CPP-603 underwater storage facility) events per year.

j. Refer to Sections 5.15.3.3 and 5.15.6.2 for details on why this frequency was not adjusted under this alternative.

k. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

Because the option exists to restart processing activities, three additional processing-related maximum reasonably foreseeable accidents are considered under this alternative (as discussed in Section 5.15.3.2). Since the amount of radioactive material that would accidentally be released to the environment from these accidents is expected to be lower than in Accidents 3 and 4 (i.e., small fuel melt and aircraft crash at the Hot Fuel Examination Facility, respectively), potential secondary impacts associated with these additional processing-related accidents would be less severe than those presented for the nonprocessing-related accidents in Table 5.15-8.

Table 5.15-12 summarizes the postulated accidents with the greatest radiological impacts under this alternative.

5.15.4.4.2.2 Alternative 4b(2) - Regionalization by Geography (Elsewhere) — Under this alternative, existing and new spent nuclear fuel inventories in the western region of the country would be centralized at either the Hanford Site or Nevada Test Site. Similar to Alternative 5a, which considers centralization of existing INEL spent nuclear fuel inventories at another DOE site, the inventory of spent nuclear fuel at the INEL would be reduced substantially so that the only spent nuclear fuel at the INEL would consist of fresh fuel generated from operating INEL reactors that had not cooled sufficiently for relocation to the regionalized or centralized site. Therefore, this alternative considers the same amount of material considered under Alternative 1 until the regionalized site could accept existing inventories of INEL spent nuclear fuel and freshly generated spent nuclear fuel that has sufficiently cooled.

Table 5.15-13 summarizes the postulated accidents with the greatest radiological impacts under this alternative.

5.15.4.5 Alternative 5: Centralization. Under this alternative, DOE would collect all current and future spent nuclear fuel inventories from both DOE and the Naval Nuclear Propulsion Program at one site. For the INEL, there are two possibilities: (1) Alternative 5a, in which most spent fuel inventories and activities would take place at the Hanford Site, Savannah River Site, Nevada Test Site, or Oak Ridge Reservation; or (2) Alternative 5b, in which all spent fuel inventories and activities would be centralized at the INEL.

5.15.4.5.1 Alternative 5a: Centralization at Other DOE Sites — This alternative would consider approximately the same amount of material considered under Alternative 1 until the centralized site could accept existing INEL spent nuclear fuel inventories and freshly generated spent

Table 5.15-12. Impacts from selected maximum reasonably foreseeable accidents - Alternative 4b(1), Regionalization by Geography (INEL) (50 and 95 percentile meteorological conditions).

Accident	Adjusted Frequency[a] (events per year)	Worker Dose[b] (rem)	Nearest Public Access[c] (rem)	Dose to MEI[d] (rem)	Offsite Population Dose (95%) (person-rem)	Adjusted point estimates of risk of fatal cancers (per year): MEI		
						MEI	Offsite Population	
						95%[e]	50%	95%
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[f]	2.0×10^{-1} (20.0)	(g)	(g)	2.0×10^{-3}	(g)	2.0×10^{-7}	(g)	(g)
2. Inadvertent criticality in ICPP[h] CPP-603 storage facility[i]	1.0×10^{-3} (1.0)[j]	9.7×10^{-2}	1.4×10^{-3}	1.0×10^{-3}	5.9×10^{-1}	5.0×10^{-10}	6.5×10^{-9} (6.5×10^{-6})[e]	3.0×10^{-7} (3.0×10^{-4})[e]
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	1.0×10^{-5} (1.0)	6.2×10^{-1}	6.5×10^{-1}	5.0×10^{0}	1.4×10^{4}	2.5×10^{-8}	4.5×10^{-7} (4.5×10^{-2})[e]	7.0×10^{-5} (7.0×10^{0})[e]
4. Material release from HFEF resulting from aircraft crash and ensuing fire	1.0×10^{-7}[(k)] (1.0)	4.6×10^{0}	3.2×10^{-1}	5.0×10^{0}	2.0×10^{3}	2.5×10^{-10}	3.6×10^{-8} (3.6×10^{-1})[e]	1.0×10^{-7} (1.0×10^{0})[e]
5. Inadvertent nuclear criticality ICPP[h] CPP-666 during processing[l]	1.0×10^{-3}	$9.1\times10^{+0}$	4.9×10^{-2}	2.8×10^{-2}	$5.6\times10^{+0}$	1.4×10^{-8}	3.1×10^{-6} (3.1×10^{-3})	2.8×10^{-6} (2.8×10^{-3})
6. Hydrogen in ICPP[h] CPP-666 dissolver	1.0×10^{-5}	(m)	(m)	6.3×10^{-4}	8.1×10^{-1}	3.2×10^{-12}	(m)	4.1×10^{-9} (4.1×10^{-4})
7. Inadvertent dissolution of 30-day cooled fuel at ICPP[h] CPP-666	1.0×10^{-6}	(m)	(m)	3.0×10^{-2}	$2.9\times10^{+1}$	1.5×10^{-11}	(m)	1.5×10^{-8} (1.5×10^{-8})

a. Numbers in parentheses indicate multiplication factor used to scale or adjust estimated accident frequencies under Alternative 1, as described in Section 5.15.3.3.

b. A worker is defined as a worker located 100 meters (328 feet) from the point of release.

c. Public individual assumed to be stranded at the nearest point of public access inside the site boundary.

d. MEI = Maximally exposed hypothetical offsite individual located at the nearest site boundary.

e. Maximally exposed individual and offsite population fatal cancer risk = dose × accident frequency × 5.0×10^{-4} fatal cancer per rem (ICRP-60 conversion factor) if dose is less than 20 rem. For doses of 20 rem or more, the ICRP-60 conversion factor is doubled, or 1.0×10^{-3}. Numbers in parentheses indicate total number of fatal cancers in the population if the accident occurs.

f. HFEF = Hot Fuel Examination Facility.

g. The safety analysis report utilized for this accident analysis does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. As demonstrated by the dose to the maximally exposed individual, consequences to the public from Accident 1 could be less than the consequences from Accidents 2 through 4. However, given the high frequency for Accident 1 compared to Accidents 2 through 4, the risk could actually be greater than for Accidents 2 through 4.

h. ICPP = Idaho Chemical Processing Plant.

i. Although three nuclear criticalities associated with spent nuclear fuel reprocessing activities have occurred during the 40-year operating history of CPP-666, the estimated frequency for an inadvertent criticality in this facility is based on existing spent nuclear conditions and fuel vulnerabilities. Nominal estimates vary from 1.0×10^{-4} (CPP-666 underwater storage facility) to 1.0×10^{-3} (CPP-603 underwater storage facility) events per year.

j. Refer to Sections 5.15.3.3 and 5.15.6.2 for details on why this frequency was not adjusted under this alternative.

k. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

l. The Idaho Chemical Processing Plant has experienced three inadvertent nuclear criticalities during its operating history, the last one 14 years ago. This frequency is based on modern facility conditions and safeguards that exist at CPP-666.

m. The safety analysis report utilized for this accident does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. However, a comparison of the data presented for this accident to the other accidents provides a relative measure of the impacts to this receptor.

Table 5.15-13. Impacts from selected maximum reasonably foreseeable accidents - Alternative 4b(2), Regionalization by Geography (Elsewhere) (50 and 95 percentile meteorological conditions).

Accident	Adjusted Frequency[a] (events per year)	Worker Dose[b] (rem)	Nearest Public Access[c] (rem)	Dose to MEI[d] (rem)	Offsite Population Dose (95%) (person-rem)	Adjusted point estimates of risk of fatal cancers (per year) MEI 95%[e]	Offsite Population 50%	Offsite Population 95%
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[f]	8.6×10^{-2} (8.6)	(g)	(g)	2.0×10^{-3}	(g)	8.6×10^{-8}	(g)	(g)
2. Inadvertent criticality in ICPP[h] CPP-603 storage facility[i]	1.0×10^{-3} (1.0)[j]	9.7×10^{-2}	1.4×10^{-3}	1.0×10^{-3}	5.9×10^{-1}	5.0×10^{-10}	6.5×10^{-9} $(6.5\times10^{-6})^e$	3.0×10^{-7} $(3.0\times10^{-4})^e$
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	1.0×10^{-5} (1.0)	6.2×10^{-1}	6.5×10^{-1}	5.0×10^{0}	1.4×10^{4}	2.5×10^{-8}	4.5×10^{-7} $(4.5\times10^{-2})^e$	7.0×10^{-5} $(7.0\times10^{0})^e$
4. Material release from HFEF resulting from aircraft crash and ensuing fire	$1.0\times10^{-7(k)}$ (1.0)	4.6×10^{0}	3.2×10^{-1}	5.0×10^{0}	2.0×10^{3}	2.5×10^{-10}	3.6×10^{-8} $(3.6\times10^{-1})^e$	1.0×10^{-7} $(1.0\times10^{0})^e$

a. Numbers in parentheses indicate multiplication factor used to scale or adjust estimated accident frequencies under Alternative 1, as described in Section 5.15.3.3.

b. A worker is defined as a worker located 100 meters (328 feet) from the point of release.

c. Public individual assumed to be stranded at the nearest point of public access inside the site boundary.

d. MEI = Maximally exposed hypothetical offsite individual located at the nearest site boundary.

e. Maximally exposed individual and offsite population fatal cancer risk = dose × accident frequency × 5.0×10^{-4} fatal cancer per rem (ICRP-60 conversion factor) if dose is less than 20 rem. For doses of 20 rem or more, the ICRP-60 conversion factor is doubled, or 1.0×10^{-3}. Numbers in parentheses indicate total number of fatal cancers in the population if the accident occurs.

f. HFEF = Hot Fuel Examination Facility.

g. The safety analysis report utilized for this accident analysis does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. As demonstrated by the dose to the maximally exposed individual, consequences to the public from this accident could be less than the consequences from Accidents 2 through 4. However, given the high frequency for this accident compared to Accidents 2 through 4, the risk could actually be greater than for Accidents 2 through 4.

h. ICPP = Idaho Chemical Processing Plant.

i. Although three nuclear criticalities associated with spent nuclear fuel reprocessing activities have occurred at the INEL during its 40-year operating history, the estimated frequency for an inadvertent criticality is not based on historic reprocessing data since reprocessing is not considered under this alternative. Nominal frequency estimates vary from 1.0×10^{-4} (CPP-666 underwater storage facility) to 1.0×10^{-3} (CPP-603 underwater storage facility) events per year.

j. Refer to Sections 5.15.3.3 and 5.15.6.2 for details on why this frequency was not adjusted under this alternative.

k. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

fuel that had cooled sufficiently. On demonstration of the centralized site's capability to receive INEL spent nuclear fuel, the inventory of spent fuel at the INEL would be reduced substantially so that the only spent nuclear fuel at the INEL would consist of fresh fuel generated from operating INEL reactors that had not cooled sufficiently for relocation to the centralized site.

Adjustments in estimated accident frequencies and point estimates of risk presented for Alternative 1 would be related to (1) the receipt, handling, and storage activities associated with the additional spent nuclear fuel inventories; and (2) the increase in overall spent fuel-related storage, relocation, and handling activities not allowed under Alternative 1. Because no changes in the accident consequences estimated for Alternative 1 are likely to occur under this alternative from increased fuel inventories (i.e., the same amount of radioactive material would accidentally be released to the environment as discussed in Section 5.15.3.3), no changes are likely in the postulated secondary impacts presented in Table 5.15-8. Table 5.15-14 summarizes the postulated accidents with the greatest radiological impacts under these alternatives.

5.15.4.5.2 Alternative 5b: Centralization at the INEL — Adjustments in estimated accident frequencies and point estimates of risk presented for Alternative 1 would be related to (1) the receipt, handling, and storage activities associated with the additional spent nuclear fuel inventories; and (2) the increase in overall spent nuclear fuel-related storage, relocation, and handling activities not allowed under Alternative 1. Because no changes in the accident consequences estimated for Alternative 1 are likely to occur under this alternative from increased fuel inventories (i.e., the same amount of radioactive material would accidentally be released to the environment as discussed in Section 5.15.3.3), no changes are likely in the postulated secondary impacts presented in Table 5.15-8. Table 5.15-15 summarizes the postulated accidents with the greatest radiological impacts under this alternative.

Because the option exists to restart processing activities, three additional processing-related maximum reasonably foreseeable accidents are considered under this alternative (as discussed in Section 5.15.3.2). Since the amount of radioactive material that would accidentally be released to the environment from these accidents is expected to be lower than Accidents 3 and 4 (i.e., small fuel melt and aircraft crash at the Hot Fuel Examination Facility, respectively), potential secondary impacts associated with these additional processing-related accidents would be less severe than those presented for the nonprocessing-related accidents in Table 5.15-8.

Table 5.15-14. Impacts from selected maximum reasonably foreseeable accidents - Alternative 5a, Centralization at Other DOE Sites (50 and 95 percentile meteorological conditions).

Accident	Adjusted Frequency[a] (events per year)	Worker Dose[b] (rem)	Nearest Public Access[c] (rem)	Dose to MEI[d] (rem)	Offsite Population Dose (95%) (person-rem)	Adjusted point estimates of risk of fatal cancers (per year)		
						MEI	Offsite Population	
						95%[e]	50%	95%
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[f]	8.6×10^{-2} (8.6)	(g)	(g)	2.0×10^{-3}	(g)	8.6×10^{-8}	(g)	(g)
2. Inadvertent criticality in ICPP[h] CPP-603 storage facility[i]	1.0×10^{-3} (1.0)[j]	9.7×10^{-2}	1.4×10^{-3}	1.0×10^{-3}	5.9×10^{-1}	5.0×10^{-10}	6.5×10^{-9} $(6.5\times10^{-6})^{e}$	3.0×10^{-7} $(3.0\times10^{-4})^{e}$
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	1.0×10^{-5} (1.0)	6.2×10^{-1}	6.5×10^{-1}	5.0×10^{0}	1.4×10^{4}	2.5×10^{-8}	4.5×10^{-7} $(4.5\times10^{-2})^{e}$	7.0×10^{-5} $(7.0\times10^{0})^{e}$
4. Material release from HFEF resulting from aircraft crash and ensuing fire	1.0×10^{-7}[(k)] (1.0)	4.6×10^{0}	3.2×10^{-1}	5.0×10^{0}	2.0×10^{3}	2.5×10^{-10}	3.6×10^{-8} $(3.6\times10^{-1})^{e}$	1.0×10^{-7} $(1.0\times10^{0})^{e}$

a. Numbers in parentheses indicate multiplication factor used to scale or adjust estimated accident frequencies under Alternative 1, as described in Section 5.15.3.3.
b. A worker is defined as a worker located 100 meters (328 feet) from the point of release.
c. Public individual assumed to be stranded at the nearest point of public access inside the site boundary.
d. MEI = Maximally exposed hypothetical offsite individual located at the nearest site boundary.
e. Maximally exposed individual and offsite population fatal cancer risk = dose × accident frequency × 5.0×10^{-4} fatal cancer per rem (ICRP-60 conversion factor) if dose is less than 20 rem. For doses of 20 rem or more, the ICRP-60 conversion factor is doubled, or 1.0×10^{-3}. Numbers in parentheses indicate total number of fatal cancers in the population if the accident occurs.
f. HFEF = Hot Fuel Examination Facility.
g. The safety analysis report utilized for this accident analysis does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. As demonstrated by the dose to the maximally exposed individual, consequences to the public from this accident could be less than the consequences from Accidents 2 through 4. However, given the high frequency for this accident compared to Accidents 2 through 4, the risk could actually be greater than for Accidents 2 through 4.
h. ICPP = Idaho Chemical Processing Plant.
i. Although three nuclear criticalities associated with spent nuclear fuel reprocessing activities have occurred at the INEL during its 40-year operating history, the estimated frequency for an inadvertent criticality is not based on historic reprocessing data since reprocessing is not considered under this alternative. Nominal frequency estimates vary from 1.0×10^{-4} (CPP-666 underwater storage facility) to 1.0×10^{-3} (CPP-603 underwater storage facility) events per year.
j. Refer to Sections 5.15.3.3 and 5.15.6.2 for details on why this frequency was not adjusted under this alternative.
k. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

Table 5.15-15. Impacts from selected maximum reasonably foreseeable accidents - Alternative 5b, Centralization at the INEL (50 and 95 percentile meteorological conditions).

Accident	Adjusted Frequency[a] (events per year)	Worker Dose[b] (rem)	Nearest Public Access[c] (rem)	Dose to MEI[d] (rem)	Offsite Population Dose (95%) (person-rem)	Adjusted point estimates of risk of fatal cancers (per year): MEI 95%[e]	Offsite Population 50%	Offsite Population 95%
1. Fuel handling accident, fuel pin breach, venting of noble gases and iodine at HFEF[f]	2.0×10^{-1} (200)	(g)	(g)	2.0×10^{-3}	(g)	2.0×10^{-7}	(g)	(g)
2. Inadvertent criticality in ICPP[h] storage facility[i]	1.0×10^{-3} (1.0)[j]	9.7×10^{-2}	1.4×10^{-3}	1.0×10^{-3}	5.9×10^{-1}	5.0×10^{-10}	6.5×10^{-9} (6.5×10^{-6})[e]	3.0×10^{-7} (3.0×10^{-4})[e]
3. Fuel melting of small number of assemblies at HFEF resulting from seismic event and cell breach	1.0×10^{-5} (1.0)	6.2×10^{-1}	6.5×10^{-1}	5.0×10^{0}	1.4×10^{4}	2.5×10^{-8}	4.5×10^{-7} (4.5×10^{-2})[e]	7.0×10^{-5} (7.0×10^{0})[e]
4. Material release from HFEF resulting from aircraft crash and ensuing fire	1.0×10^{-7}[(k)] (1.0)	4.6×10^{0}	3.2×10^{-1}	5.0×10^{0}	2.0×10^{3}	2.5×10^{-10}	3.6×10^{-8} (3.6×10^{-1})[e]	1.0×10^{-7} (1.0×10^{0})[e]
5. Inadvertent nuclear criticality ICPP[h] CPP-666 during processing[l]	1.0×10^{-3}	$9.1\times10^{+0}$	4.9×10^{-2}	2.8×10^{-2}	$5.6\times10^{+0}$	1.4×10^{-8}	3.1×10^{-6} (3.1×10^{-3})	2.8×10^{-6} (2.8×10^{-3})
6. Hydrogen in ICPP[h] CPP-666 dissolver	1.0×10^{-5}	(m)	(m)	6.3×10^{-4}	8.1×10^{-1}	3.2×10^{-12}	(m)	4.1×10^{-9} (4.1×10^{-4})
7. Inadvertent dissolution of 30-day cooled fuel at ICPP[h] CPP-666	1.0×10^{-6}	(m)	(m)	3.0×10^{-2}	$2.9\times10^{+1}$	1.5×10^{-11}	(m)	1.5×10^{-8} (1.5×10^{-2})

a. Numbers in parentheses indicate multiplication factor used to scale or adjust estimated accident frequencies under Alternative 1, as described in Section 5.15.3.3.

b. A worker is defined as a worker located 100 meters (328 feet) from the point of release.

c. Public individual assumed to be stranded at the nearest point of public access inside the site boundary.

d. MEI = Maximally exposed hypothetical offsite individual located at the nearest site boundary.

e. Maximally exposed individual and offsite population fatal cancer risk = dose × accident frequency × 5.0×10^{-4} fatal cancer per rem (ICRP-60 conversion factor) if dose is less than 20 rem. For doses of 20 rem or more, the ICRP-60 conversion factor is doubled, or 1.0×10^{-3}. Numbers in parentheses indicate total number of fatal cancers in the population if the accident occurs.

f. HFEF = Hot Fuel Examination Facility.

g. The safety analysis report utilized for this accident analysis does not provide this information because it was developed prior to DOE Orders requiring this information. As demonstrated by the dose to the maximally exposed individual, consequences to the public from this accident could be less than the consequences from Accidents 2 through 4. However, given the high frequency for this accident compared to Accidents 2 through 4, the risk could actually be greater than for Accidents 2 through 4.

h. ICPP = Idaho Chemical Processing Plant.

i. Although three nuclear criticalities associated with spent nuclear fuel reprocessing activities have occurred during the 40-year operating history of CPP-666, the estimated frequency for an inadvertent criticality in this facility is based on existing spent nuclear conditions and fuel vulnerabilities. Nominal estimates vary from 1.0×10^{-4} (CPP-666 underwater storage facility) to 1.0×10^{-3} (CPP-603 underwater storage facility) events per year.

j. Refer to Sections 5.15.3.3 and 5.15.6.2 for details on why this frequency was not adjusted under this alternative.

k. This frequency is a qualitative bounding estimate for a potential aircraft crash, as discussed in Section 5.15.6.4.

l. The Idaho Chemical Processing Plant has experienced three inadvertent nuclear criticalities during its operating history, the last one 14 years ago. This frequency is based on modern facility conditions and safeguards that exist at CPP-666.

m. The safety analysis report utilized for this accident does not provide this information because it was developed prior to DOE Order 5480.23 requiring this information. However, a comparison of the data presented for this accident to the other accidents provides a relative measure of the impacts to this receptor.

5.15.5 Impacts from Postulated Maximum Reasonably Foreseeable Toxic Material Accidents

Like radioactive materials, toxic materials (e.g., chemicals) are involved in a variety of operations, including spent nuclear fuel-related activities, at the INEL. As a result of these operations and activities, the potential exists for releases of toxic materials to the environment from the same types of initiators considered in determining the radiological accident scenarios discussed in Section 5.15.4. This section summarizes analyses of postulated accident scenarios associated with spent nuclear fuel activities that could result in the release of toxic materials from their confinements.

5.15.5.1 Identification of Toxic Chemicals at the INEL. The facilities at the INEL use many types and quantities of chemically toxic materials. To determine the spent fuel-related chemicals that exist in sufficient quantities to present health effects to workers or the offsite population, DOE performed an initial screening of the chemical inventories at the INEL. This screening consisted of identifying those hazardous chemicals at the INEL listed in the Superfund Amendments and Reauthorization Act of 1986 (SARA) 312 Report for 1992 (Priestly 1992) that (1) exist in bulk quantities [assumed to be greater than 227 kilograms (500 pounds)]; or (2) exceed reportable quantities [usually 0.45 kilogram (1 pound)] on the EPA Title III List of Lists (EPA 1990), which includes hazardous chemicals defined in the following:

- SARA Section 302, Extremely Hazardous Substances (40 CFR Part 355, Appendixes A and B, List of Extremely Hazardous Substances and Their Threshold Planning Quantities) (CFR 1993)

- Comprehensive Environmental Response, Compensation, and Liability Act Hazardous Substances (40 CFR Part 302, Table 302.4, Lists of Hazardous Substances and Reportable Quantities) (CFR 1992a)

- SARA Section 313, Toxic Chemicals (CFR 1992b)

- Federal Register list of 100 extremely hazardous chemicals (FR 1994)

5.15.5.2 Selection of Spent Nuclear Fuel-Related Toxic Chemicals Requiring Accident Analysis. As indicated by the screening methodology discussed above, toxic chemical inventories are located throughout INEL facilities in varying quantities and are involved in nearly all operations and activities performed by INEL facilities, including spent nuclear fuel-related activities.

The screening identified no toxic chemicals associated with the dry storage of spent nuclear fuel. Except for processing-related activities that could be performed under the Regionalization and Centralization at INEL alternatives [i.e., Alternatives 4b(1) and 5b, respectively], the screening identified activities associated with the underwater storage of spent nuclear fuel (e.g., maintaining water chemistry) as the only spent nuclear-fuel related activities that might utilize toxic chemicals in sufficient quantities to present a potential for health effects to workers or the offsite population, or potential contamination of the environment. For Alternatives 4b(2) and 5a, in which DOE would relocate INEL spent nuclear fuel inventories and related activities to other DOE sites, the existing toxic chemical inventories at the INEL would be expected to slightly decrease. For Alternatives 4b(1) and 5b, in which the INEL could potentially resume processing activities, a substantial increase in existing chemical inventories, primarily hydrofluoric acid and anhydrous ammonia, would be expected. No substantial changes in existing spent nuclear fuel-related toxic chemical inventories would be expected under Alternatives 1, 2, or 3.

To demonstrate how the consequences of the same accident at an identical hypothetical facility constructed at the Hanford Site or the Savannah River Site under this alternative would compare to the INEL (based on local geological and meteorological conditions), Appendix D summarizes postulated accident scenarios for a new Expended Core Facility that DOE could construct at any of the sites considered in this EIS.

To determine potential accident scenarios associated with handling or storing toxic chemicals at the various spent nuclear fuel-related facilities, DOE performed an extensive review of existing safety analyses and walkdowns of various facilities. This review identified two nonprocessing-related toxic chemicals at the Idaho Chemical Processing Plant — nitric acid and chlorine — as requiring further evaluation to determine potential health effects to workers and the offsite population. Additionally, two toxic chemicals that would be required to support the resumption of processing activities at the Idaho Chemical Processing Plant — hydrofluoric acid and anhydrous ammonia — were identified as requiring further evaluation.[6] Although spent fuel-related facilities at the Idaho Chemical Processing Plant use several other toxic chemicals (e.g., oxalic acid), the quantities of these chemicals are not sufficient to present an impact to workers or the environment from accidental releases to the

6 Although bulk quantities of nitric acid would be required to perform processing activities that could be resumed under Alternatives 4b(1) and 5b, the consequences of processing-related accidents involving nitric acid would be bounded by the hydrofluoric acid and anhydrous accidents analyzed in Sections 5.15.5.3.3 and 5.15.5.3.4, respectively. Therefore, this analysis focuses on a potential nitric acid accident resulting from the nonprocessing spent nuclear fuel-related activities considered under the other alternatives.

environment. (For postulated accident scenarios involving Naval spent nuclear fuel-related activities at the INEL, refer to Appendix D.)

Because DOE determined that it needed to evaluate postulated toxic chemical accidents at the Idaho Chemical Processing Plant as part of this EIS, it did not consider postulated toxic chemical accidents at the Advanced Test Reactor Storage Canal and the Hot Fuel Examination Facility that could be involved in spent fuel-related activities[7] for further evaluation in this EIS for the following reasons:

- In general, quantities of spent nuclear fuel-related chemicals at the Idaho Chemical Processing Plant are substantially greater than those at the Advanced Test Reactor Storage Canal and Hot Fuel Examination Facility.

- The Idaho Chemical Processing Plant is located approximately 1,000 meters (1,094 yards) closer to the nearest site boundary than the Advanced Test Reactor.

Based on a review of safety documentation for the Test Area North spent nuclear fuel underwater storage facility and discussions with facility personnel, DOE determined that none of the toxic chemicals identified in the screening (Section 5.15.5.1) is related to spent fuel handling or storage activities.

5.15.5.3 Toxic Chemical Accident Analysis. For chemically toxic materials, several government agencies recommend quantifying health effects that cause short-term effects as threshold values of concentrations in air or water. The long-term health consequences of human exposure to toxic materials are not as well understood as the long-term health consequences related to radiation exposure. Thus, the potential health effects for exposures to toxic chemicals are more subjective than those for radioactive materials. Factors such as receptor locations, terrain, meteorological conditions, release conditions, and characteristics of chemical inventories are required parameters for determinations of airborne concentrations of toxic chemicals at various distances from a postulated point of release.

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7 The scope of this analysis has been restricted to the Advanced Test Reactor fuel storage canal. Everything inside the reactor gas-tight boundary and associated with reactor operations has been excluded from consideration because reactor operations are not related to the spent nuclear fuel activities considered in this EIS.

EPICode™ was used to estimate airborne concentrations resulting from spent nuclear fuel-related toxic chemical releases at the INEL. [For a detailed description of EPICode™, refer to Slaughterbeck et al. (1995).]

To determine the potential health effects from accidental releases of toxic chemicals, this analysis compared the concentrations determined by EPICode™ against Emergency Response Planning Guideline values, where available. These values, which are specific for each substance, are related to three general severity levels:

- Exposure to concentrations greater than Emergency Response Planning Guideline-1 values for a period of time greater than 1 hour results in an unacceptable likelihood that a person would experience mild transient adverse health effects, or perception of a clearly defined objectionable odor.

- Exposure to concentrations greater than Emergency Response Planning Guideline-2 values for a period of time greater than 1 hour results in an unacceptable likelihood that a person would experience or develop irreversible or other serious health effects, or symptoms that could impair one's ability to take protective action.

- Exposure to concentrations greater than Emergency Response Planning Guideline-3 values for a period of time greater than 1 hour results in an unacceptable likelihood that a person would experience or develop life-threatening health effects.

If there were no Emergency Response Planning Guideline values for a toxic substance, the analysis substituted other chemical toxicity values, as follows:

- Threshold limit values/time-weighted average values (ACGIH 1988) substituted for Emergency Response Planning Guideline-1. This is the time-weighted average concentration for a normal 8-hour workday and a 40-hour workweek to which nearly all workers could be repeatedly exposed, day after day, without adverse effect.

- Level of concern values (equal to 0.1 of the immediately dangerous to life or health values - see below) substituted for Emergency Response Planning Guideline-2. The level of concern value is the concentration of a hazardous substance in the air above which there might be serious irreversible health effects or death as a result of a single exposure for a relatively short period of time.

- Immediately dangerous to life or health values are substituted for Emergency Response Planning Guideline-3. The immediately dangerous to life or health value is the maximum concentration from which a person could escape within 30 minutes without a respirator and without experiencing any impairment of escape or irreversible side effects (NIOSH 1990).

As stated in the above section, four toxic chemicals — chlorine, nitric acid, hydrofluoric acid, and anhydrous ammonia — at the Idaho Chemical Processing Plant were identified as requiring further evaluation to estimate potential health effects to workers and the public. The following sections summarize the analyses performed for these chemicals.

5.15.5.3.1 Accidental Chlorine Release — Chlorine, while not directly associated with spent nuclear fuel-related activities at the INEL, is used to treat drinking water supplies at the various spent fuel facilities. Therefore, an analysis of a postulated accidental chlorine release at the Idaho Chemical Processing Plant was performed to determine potential impacts on workers operating the spent fuel-related facilities.

At the Idaho Chemical Processing Plant, chlorine is contained in two pressurized bottles [65 atmospheres at 20°C (68°F)], a 68-kilogram (150-pound) bottle and a 55-kilogram (120-pound) bottle, totaling 123 kilograms (270 pounds). To be conservative, DOE assumed that a breach of the drain line causes an instantaneous release of the total inventory of both tanks. The highest chlorine concentrations at the receptor locations would result from the largest release over the shortest time period. Therefore, the release duration was assumed to be approximately 5 minutes.

An accidental chlorine release from one of the chlorine tanks could be initiated by one of several events, such as a handling event, piping or valve rupture, or human error. Because the two tanks are physically separated, an accidental simultaneous release from both tanks would require a common initiator such as a delivery accident, a common maintenance failure, or a natural phenomena event (e.g., seismic) that damaged or punctured both tanks. The frequency of an accidental release from one pressurized tank is 1.0×10^{-4} event per year (EPA/FEMA/DOT 1987). A common cause failure resulting in the release of chlorine from two separated tanks is assumed to be no greater than 5 percent of the time given for the first tank failure. Therefore, the estimated frequency of an accidental release from both tanks is 5.0×10^{-6} events per year (with no credit taken for pressure vessel management and training).

Table 5.15-16 summarizes the concentrations of the subject chlorine release at the following receptor locations: a facility worker, a member of the public stranded at the nearest point of public access inside the INEL boundary, and a maximally exposed hypothetical member of the public located at the nearest site boundary. As listed in Table 5.15-10, the peak chlorine concentrations for facility workers could result in life-threatening health effects (i.e., Emergency Response Planning Guideline-3 values are exceeded) for both conservative (95 percentile) and average (50 percentile) meteorological conditions.

Table 5.15-16. Summary of chemical concentrations for postulated nonprocessing-related accidental releases at the Idaho Chemical Processing Plant under Alternatives 1 through 5.

	Chemical Concentrations (milligrams per cubic meter)[a]			
	95% Meteorology[b]		50% Meteorology[c]	
Receptor Location	Chlorine ERPG-1[d] = 3 (1) ERPG-2 = 9 (3) ERPG-3 = 60 (20)	Nitric Acid[e] TWA = 5.2 (2) LOC = 25.5 (10) IDLH = 255 (100)	Chlorine ERPG-1 = 3 (1) ERPG-2 = 9 (3) ERPG-3 = 60 (20)	Nitric Acid[e] TWA = 5.2 (2) LOC = 25.5 (10) IDLH = 255 (100)
1. Worker located at 100 meters (325 feet).	84,000 (28,000)	250 (95)	1,620 (540)	33 (13)
2. Nearest point of public access where a member of the public is assumed stranded at the time of the release.[f]	19.5 (6.5)	0.32 (0.12)	1.89 (0 63)	0.049 (0.019)
3. Maximally exposed hypothetical individual located at the nearest site boundary.[g]	4.2 (1.4)	0.12 (0.047)	0.42 (0.14)	0.016 (0.006)

a. Numbers in parentheses reflect concentrations in parts per million.
b. The 95 percentile meteorology is based on Class F (unfavorable) meteorological conditions with 0.5 meter per second (1.1 miles per hour) wind speed for receptors located within 2 kilometers (1.2 miles) of the release and 2 meters per second (4.5 miles per hour) for receptors beyond 2 kilometers of the release.
c. The 50 percentile meteorology is based on Class D (typical) meteorological conditions with 4.5 meters per second (10 miles per hour) wind speed for all receptors.
d. ERPG = Emergency Response Planning Guidelines.
e. Because Emergency Response Planning Guideline values are not available for nitric acid, time-weighted average values are substituted for ERPG-1 values, level of concern values are substituted for ERPG-2 values, and immediately dangerous to life or health values are substituted for Emergency Response Planning Guideline-3 values. Refer to Section 5.15.5.3 for further information regarding the use of these values.
f. The nearest point of public access from this postulated release is 5,870 meters (6,419 yards).
g. The nearest site boundary is located at 14,000 meters (15,310 yards).

Peak chlorine concentrations estimated at the nearest point of public access can exceed the Emergency Response Planning Guideline-2 value assuming 95 percentile meteorological conditions, as listed in Table 5.15-10. Symptoms associated with exposure to these concentrations could include burning of the eyes, nose, and throat, coughing, choking, and possibly skin burns.

As listed in Table 5.15-16, the estimated peak averaged chlorine concentration at the nearest site boundary would be above the Emergency Response Planning Guideline-1 value for 95 percentile meteorological conditions. However, due to the nature of the release, this concentration probably would not last for more than a few minutes. Therefore, it would be likely that individuals at this distance would experience no more than mild transient adverse health effects.

This analysis took limited credit for emergency response actions following a chlorine release in calculating the concentrations listed in Table 5.15-16. To mitigate the consequences of a chlorine release to the environment, the same emergency response programs and actions described for radiological accident scenarios (Section 5.15.4.1) would be initiated following the release. Therefore, actual health effects experienced by persons inside the site boundary would realistically be less than the values listed in Table 5.15-16.

Because the estimated airborne concentration of chlorine at 100 meters (328 feet) substantially exceeds the guidelines listed in Table 5.15-16, workers could be fatally injured or could receive long-term or permanent health effects. Potential secondary impacts associated with the chlorine accident scenario would involve economic impacts such as workers' compensation, medical bills, and potential lawsuits. No other secondary impacts, such as impacts on national defense or biotic resources, were identified.

5.15.5.3.2 Accidental Nitric Acid Release — Nitric acid is used at various spent nuclear fuel-related storage facilities for maintaining the chemistry of the water used in underwater storage facilities.[8] Based on the toxic chemical screening discussed in Section 5.15.5.1, review of existing safety analyses, walkdowns of spent nuclear fuel-related facilities, and interviews with INEL

8 Although bulk quantities of nitric acid would be required to perform processing activities that could be resumed under Alternatives 4b(1) and 5b, the consequences of processing-related accidents involving nitric acid would be bounded by the hydrofluoric acid and anhydrous accidents analyzed in Sections 5.15.5.3.3 and 5.15.5.3.4, respectively. Therefore, this analysis focuses on a potential nitric acid accident resulting from the non-processing spent nuclear fuel-related activities considered under the other alternatives.

personnel, DOE determined that the potential exists for an accidental release of nitric acid from one of two 1,135 liters (300-gallon) storage tanks used to support spent nuclear fuel-related water treatment activities at the Idaho Chemical Processing Plant. Because one of the tanks is usually empty, the two tanks have separate valves, and they are physically separated, DOE could not identify a reasonably likely initiator that could cause an accidental simultaneous release from both tanks.

The quantity of nitric acid assumed available for release from a single initiator would be (1,135 liters) 300 gallons. The following assumptions were made for this analysis:

- An initiating event causes severe structural damage (e.g., large puncture) to one of the tanks.
- The entire inventory of nitric acid is released into the containment wall surrounding the storage tank.
- The area of the containment wall is approximately 28 square meters (300 square feet).
- The total release of nitric acid [i.e., 1.135 liters (300 gallons)] evaporates into the atmosphere before the implementation of emergency response procedures can recover the nitric acid.

Table 5.15-16 summarizes the concentrations of the nitric acid release at the following receptor locations for both conservative (95 percentile) and average (50 percentile) meteorological conditions: a facility worker, a member of the public stranded at the nearest point of public access inside the INEL boundary, and a maximally exposed hypothetical member of the public at the nearest site boundary. The estimated frequency for this event is 1×10^{-5} events per year.

This analysis took limited credit for emergency response actions following a nitric acid release in calculating the concentrations listed in Table 5.15-16. To mitigate the consequences of a release to the environment, the same emergency response programs and actions described for radiological accident scenarios (Section 5.15.4.1) would be initiated following a nitric acid release. Therefore, actual health effects experienced by persons inside the site boundary would realistically be less than the values listed in Table 5.15-16.

Other than limited economic secondary impacts, no other secondary impacts would be likely if this accident occurred.

5.15.5.3.3 Accidental Hydrofluoric Acid Release — To resume spent nuclear fuel processing activities at the Fluorinel and Storage (FAST) facility (CPP-666), which is currently shutdown and being placed in a permanent shutdown mode, bulk quantities of hydrofluoric acid would be required to support the dissolution process. A hydrofluoric acid storage tank with an operating capacity of approximately 30,283 liters (8,000 gallons) is located in the Idaho Chemical Processing Plant facility area to support processing activities, although only 11,356 liters (3,000 gallons) of hydrofluoric acid remain in the tank, and efforts are currently underway to remove the remaining hydrofluoric acid in the tank from the INEL site.

Table 5.15-17 summarizes the potential impacts upon a maximally exposed hypothetically offsite individual located at the nearest site boundary [14,000 meters (15,310 yards)] resulting from a potential hydrofluoric acid release at the Idaho Chemical Processing Plant assuming 95 percentile meteorological conditions. Slaughterbeck et al. (1995) provides further details and discussion regarding this postulated accident scenario. Although Slaughterbeck et al. (1995) presents impacts to only the maximally exposed offsite hypothetical individual resulting from this postulated accident for 95 percentile meteorological conditions, a comparison of the airborne concentration of hydrofluoric acid at 14,000 meters (15,310 yards) to the airborne concentrations from other postulated chemical accident scenarios (as presented in Table 5.15-16) at the same receptor distance provides meaningful perspective on the significance of this accident.

Table 5.15-17. Summary of chemical concentrations for postulated processing-related accidental releases at the Idaho Chemical Processing Plant under Alternatives 4b(1) and 5b.

	Chemical Concentrations (milligrams per cubic meter)[a]	
	95% Meteorology[b]	
Receptor Location	Hydrofluoric Acid ERPG-1[c] = 4 (5) ERPG-2 = 17 (20) ERPG-3 = 43 (50)	Anhydrous Ammonia ERPG-1 = 17 (25) ERPG-2 = 136 (200) ERPG-3 = 680 (1000)
Maximally exposed hypothetical individual located at the nearest boundary[d]	0.078 (0.09)	82 (120.6)

a. Numbers in parentheses reflect concentrations in parts per million.
b. The 95 percentile meteorology is based on Class F (unfavorable) meteorological conditions with 0.5 meter per second (1.1 miles per hour) wind speed for receptors located within 2 kilometers (1.2 miles) of the release and 2 meters per second (4.5 miles per hour) for receptors beyond 2 kilometers of the release.
c. ERPG = Emergency Response Planning Guidelines.
d. The nearest site boundary is located at 14,000 meters (15,310 yards).

The estimated frequency for this event is 1×10^{-5} events per year. It should be noted that this potential accident applies only to Alternatives 4b(1) and 5b, and is in addition to the potential chlorine and nitric acid release accidents described in Sections 5.15.5.3.1 and 5.15.5.3.2, respectively.

This analysis took limited credit for emergency response actions following a hydrofluoric acid release in calculating the concentrations listed in Table 5.15-17. To mitigate the consequences of a release to the environment, the same emergency response programs and actions described for radiological accident scenarios (Section 5.15.4.1) would be initiated following a hydrofluoric acid release. Therefore, actual health effects experienced by persons inside the site boundary would realistically be less than the values listed in Table 5.15-17.

Other than limited economic secondary impacts, no other secondary impacts would be likely if this accident occurred.

5.15.5.3.4 Accidental Anhydrous Ammonia Release — To resume spent nuclear fuel processing activities at the Fluorinel and Storage (FAST) facility (CPP-666), bulk quantities of anhydrous ammonia would be required to support operation of the NO_x-Abatement Facility (CPP-1670), a facility that would be constructed to treat airborne effluents from the INEL processing facilities before being released to the environment.

The NO_x-Abatement Facility would be expected to utilize two anhydrous ammonia tanks, each with a storage capacity of 68,000 liters (18,000 gallons). Table 5.15-17 summarizes the potential impacts upon the maximally exposed hypothetical offsite individual located at the nearest site boundary [14,000 meters (15,310 yards)] resulting from a short-term release of the contents of both storage tanks [i.e., 136,000 liters (36,000 gallons)] at the Idaho Chemical Processing Plant assuming 95 percentile meteorological conditions. Slaughterbeck et al. (1995) provides further details and discussion regarding this postulated accident scenario. Although Slaughterbeck et al. (1995) presents only impacts to the maximally exposed offsite hypothetical individual resulting from this postulated accident for 95 percentile meteorological conditions, a comparison of the airborne concentration of anhydrous ammonia at 14,000 meters (15,310 yards) to the airborne concentrations from other postulated chemical accident scenarios (as presented in Table 5.15-16) at the same distance provides meaningful perspective on the significance of this accident.

The estimated frequency for this event is 5×10^{-6} events per year. The basis for this estimated frequency is identical to that described for an accidental chlorine release from two separate tanks, as described in Section 5.15.5.3.1. It should be noted that this potential accident applies only to Alternatives 4b(1) and 5b, and is in addition to the potential chlorine and nitric acid release accidents described in Sections 5.15.5.3.1 and 5.15.5.3.2, respectively.

This analysis took limited credit for emergency response actions following an anhydrous ammonia release in calculating the concentrations listed in Table 5.15-17. To mitigate the consequences of a release to the environment, the same emergency response programs and actions described for radiological accident scenarios (Section 5.15.4.1) would be initiated following a hydrofluoric acid release. Therefore, actual health effects experienced by persons inside the site boundary would realistically be less than the values listed in Table 5.15-17.

Other than limited economic secondary impacts, no other secondary impacts would be likely if this accident occurred.

5.15.6 Maximum Reasonably Foreseeable Radiological Accident Scenario Descriptions

The purpose of this section is to summarize the different accident scenarios identified in Section 5.15.4. The Facility Safety Report for the Argonne National Laboratory-West Hot Fuel Examination Facility (ANL 1975) contains further details and discussions for Accident 1, discussed below. Slaughterbeck et al. (1995) provides further details, discussions, and references for Accidents 2 through 7, discussed below. Additional discussions and references regarding the processing-related accidents summarized in this section are also provided in a study performed to determine the potential impacts spent nuclear fuel processing-related accidents could have on the siting of a new production reactor at the INEL (EG&G 1993b). These documents contain additional information, such as release fractions, source terms, and other assumptions used in the accident analyses. Appendix D describes postulated accident scenarios associated with Naval spent nuclear fuel-related facilities and activities at the INEL.

5.15.6.1 Accident 1: Fuel Pin Breach and Venting of Noble Gases and Iodine to the Environment from a Mechanical Handling Accident at the Argonne National Laboratory-West Hot Fuel Examination Facility. The accident screening methodology discussed in Section 5.15.3 identified a mechanical handling event at the Argonne National Laboratory-West Hot Fuel Examination Facility as an initiator to the maximum reasonably foreseeable accident within the abnormal event frequency range. This event would result in a fuel pin breach and venting of noble gases and iodine to the environment. The identification of this accident as a maximum reasonably foreseeable accident is based on the estimated radiological consequences to the maximally exposed hypothetical offsite individual at the nearest site boundary presented in the Hot Fuel Examination Facility Safety Report (ANL 1975). Othcr postulated accidents associated with handling spent nuclear fuel in the Hot Fuel Examination Facility before the identification of the fuel pin breach accident as the maximum reasonably foreseeable accident included an inadvertent criticality and a sodium fire. A fuel pin breach accident was chosen as the maximum reasonably foreseeable accident because the estimated frequencies for an inadvertent criticality and a sodium fire in the facility are extremely low (ANL 1975).

The analyses defined in the Facility Safety Report (ANL 1975) made the following assumptions:

- The fuel subassemblies and experimental capsules being examined in the facility were cooled for at least 15 days to ensure that the short-lived fission products had decayed.

- The noble gases and iodines that could be released from this accident scenario were immediately released.

- One hundred percent of the noble gases, 25 percent of the iodines, and 1 percent of particulates were available for escape to the atmosphere.

- The building containment structure, including the building ventilation system, and the Main Cell, including the argon ventilation system, remained operational following the handling accident. This assumption is considered appropriate because the mechanical handling accident scenario under consideration would not initiate a failure in these systems. (Accident 3 considers the simultaneous failure of all these systems in conjunction with the melting of fuel assemblies stored in the facility).

The Facility Safety Report (ANL 1975) contains specific information on the source terms associated with breaching the fuel section of a pin. Because that report does not provide an estimated frequency of occurrence for the subject mechanical handling accident scenario, the analysis used historic information and engineering judgment to determine the conservatively estimated frequency for this accident of 1.0×10^{-2} event per year.

For determining the impacts from this postulated accident scenario, the nearest point of public access is equivalent to the nearest site boundary, which is 5,240 meters (5,730 yards) from the point of the release. Although the Facility Safety Report (ANL 1975) does not estimate consequences to the offsite population resulting from this accident scenario, this analysis reasonably estimated that the exposures (i.e., dose) to the offsite population would be less than the offsite population dose calculated for Accidents 2 through 4 because the dose to the maximally exposed hypothetical individual at the nearest site boundary from this accident would be less than that estimated for Accidents 2 through 4.

5.15.6.2 Accident 2: Inadvertent Nuclear Chain Reaction in Wet Spent Nuclear Fuel Storage (1×10^{19} fissions, 8-hour release) at the Idaho Chemical Processing Plant CPP-603 Underwater Fuel Storage Facility. The accident screening methodology discussed in Section 5.15.3 identified an inadvertent nuclear criticality associated with underwater spent nuclear fuel storage at the CPP-603 Underwater Fuel Storage Facility as an accident requiring further evaluation. Other postulated accidents that were considered before the identification of an inadvertent criticality accident as a maximum reasonably foreseeable accident included pool leaks, fuel damage events, and loss of cooling events. This analysis selected an inadvertent nuclear criticality for evaluation in this EIS over the other accidents for the following reasons:

- Postulated inadvertent nuclear criticality accidents have been addressed in virtually all DOE nonreactor EISs and safety analysis reports in which such accidents were reasonably foreseeable because of public concerns regarding the potential for these accidents.

- The Idaho Chemical Processing Plant has experienced three inadvertent nuclear criticality accidents. Although none of these accidents involved a fuel storage facility, they demonstrate the potential and concern for such events.

- The consequences of water leakage from a pool-draining event would present lower prompt consequences to workers than a criticality because the INEL could implement emergency response plans to evacuate workers before the risk to these workers could substantially increase. In addition, a pool drain was considered to be an initiator to a criticality accident.

- Mechanical fuel damage events are less impacting than a nuclear chain reaction scenario because some degree of fuel damage is part of the criticality accident scenario and analysis.

Of the different Idaho Chemical Processing Plant facility areas that store spent nuclear fuel, the CPP-603 Underwater Fuel Storage Facility was selected for analysis of a criticality accident for the following reasons:

- CPP-603 facility storage includes most types of spent nuclear fuel stored elsewhere on the site. Fuel stored at reactor basins is an exception (but was considered in the determination of other reasonably foreseeable accident scenarios) because of its much shorter cooling times after removal from a reactor.

- CPP-603 facility spent nuclear fuel storage quantities are comparable to or exceed the spent nuclear fuel inventories stored elsewhere on the site.

- The CPP-603 facility is an older facility that does not contain all the preventive or mitigative design features found in more modern facilities, such as the CPP-666 Fuel Storage Area.

The analysis selected the underwater fuel storage portion of the CPP-603 facility rather than the Irradiated Fuels Storage Facility portion of the CPP-603 facility because accidents involving graphite fuels in dry storage probably would have less severe potential consequences because they had been removed from reactors for a much longer period of time and, because of their design, would prevent most of the remaining fission products from being released if a criticality accident occurred.

Initiating events that the analysis considered possible to lead to an inadvertent nuclear criticality included operator error, hanger corrosion, equipment failure, an earthquake, pool drain, and an aircraft crash. The scenario discussed in this EIS assumes a postulated criticality scenario that could be initiated by human error, equipment failure, or earthquake. Heat generated from the chain reaction would easily dissipate and thereby avoid fuel melting but would still cause the release of fission products associated with 1×10^{19} fissions over an 8-hour period.

Between 1945 and 1980, 40 known inadvertent criticalities occurred worldwide, none of which involved the handling or storage of spent nuclear fuel in an underwater fuel storage facilities. In addition, between 1975 and 1980, there were 160 nuclear power reactor facilities with underwater fuel storage facilities worldwide. None of these facilities ever had a nuclear criticality associated with its underwater storage facilities. Therefore, it is generally assumed that the likelihood for such an event in a modern underwater storage facility is unlikely, with a frequency estimated at 1×10^{-4} event per year. This estimated frequency is supported by information in the safety analysis report for the CPP-666 underwater storage facility, which is a modem facility (e.g., 1980s vintage) at the INEL used to store various types of spent nuclear fuel. In the CPP-603 Underwater Fuel Storage Facility, however, where spent nuclear fuel inventories have substantially corroded or degraded (DOE 1993c), and where the design of the facility and its supporting equipment do not meet current design specifications, activities associated with handling and storing spent nuclear fuel present an increase in the likelihood for an inadvertent nuclear criticality accident by as much as an order of magnitude. Therefore, this analysis conservatively assumes the estimated frequency for an inadvertent nuclear criticality associated with handling spent nuclear fuel in the CPP-603 Underwater Fuel Storage Facility to be 1×10^{-3} event per year for this analysis.

The handling activities associated with stabilizing CPP-603 facility spent nuclear fuel inventories would occur under each of the five alternatives considered in this EIS. The estimated frequency for an inadvertent criticality at the CPP-603 facility is an order of magnitude larger than that of any other INEL facility (e.g., 1×10^{-3} event per year), and is considered a "worst-case" frequency that bounds changes in estimated criticality frequencies at other INEL facilities resulting from increased handling activities associated with changes in spent nuclear fuel inventories. Therefore, using the estimated criticality frequency related to the CPP-603 as the estimated frequency under each alternative provides a conservative bound on the estimated criticality frequencies for other spent nuclear fuel-related handling and storage facilities.

To determine the accident impacts from this postulated accident scenario, the analysis assumed the worker to be located 100 meters (328 feet) from the event, the nearest point of public access (U.S. Route 20/26) is 5,870 meters (6,420 yards), and the nearest site boundary is located at 14,000 meters (15,310 yards).

5.15.6.3 Accident 3: Earthquake-Induced Breach and Fuel Melt at the Argonne National Laboratory-West Hot Fuel Examination Facility. The accident screening methodology discussed in Section 5.15.3 identified an earthquake-induced breach and fuel melt at the Argonne National Laboratory-West Hot Fuel Examination Facility as a maximum reasonably foreseeable accident that would present higher radiological consequences to facility workers or the offsite population than other postulated accidents analyzed in the same accident frequency range. The postulated events leading to atmospheric release of radionuclides are as follows:

- The earthquake results in a peak horizontal ground acceleration of sufficient magnitude to cause structural damage to the building structure and a large breach in the main cell.[9]

- Coincident with the breach, a failure of the fuel subassembly cooling system occurs, resulting in the melting of fresh assemblies.

- Radionuclides from the melting fuel subassemblies are released to the atmosphere.

The estimated probability of an earthquake in the Argonne National Laboratory-West facility area resulting in a peak horizontal acceleration of sufficient magnitude to damage the facility structure and breach the cell is 1×10^{-5} event per year. This analysis conservatively assumes the probability of failure of the building structure, Main Cell, and subassembly cooling to be 1.0, given that the earthquake has occurred. A preliminary assessment of the seismic integrity of the Hot Fuel Examination Facility, as discussed in Slaughterbeck et al. (1995), indicates that, given the current state of analysis, significant failures could result at the Hot Fuel Examination Facility from this earthquake.

In determining the number of fuel assemblies that would be affected during this scenario, the analysis assumed that 20 fuel subassemblies would melt due to failure of the forced cooling in this accident. Although 40 storage positions are available for fuel that would require forced cooling, current plans do not estimate the need to use more than 20 of these positions. The release duration for this scenario is 30 days. To prevent doses greater than 5 rem to the public from this scenario, the analysis assumed intervention by evacuation or prevention of contaminated food consumption, with the calculated doses reflecting this assumption.

To determine the impacts from this postulated accident scenario, the analysis assumed the worker to be located 100 meters (328 feet) from the event, and the nearest point of public access (U.S. Route 20) and the nearest site boundary at 5,240 meters (5,730 yards).

9 As discussed in Slaughterbeck et al. (1995), accelerations with any of several potential seismic events with a combined estimated frequency of 1×10^{-5} per year are beyond the design of the Hot Fuel Examination Facility and were determined to compromise the ability of the structure to maintain confinement. Events this rare are beyond the requirements of DOE Order 5480.28 and DOE-ID Architectural Engineering Standards for Category 1 (high hazard) facilities.

5.15.6.4 Accident 4: Radiological Material Release from the Argonne National Laboratory-West Hot Fuel Examination Facility Resulting from an Aircraft Crash and Ensuing Fire. The accident screening methodology discussed in Section 5.15.3 identified a radioactive material release from the Argonne National Laboratory-West Hot Fuel Examination Facility resulting from an aircraft crash as the maximum reasonably foreseeable accident in the beyond-design-basis accident frequency range. Of externally initiated events, an aircraft crash into the Hot Fuel Examination Facility is a maximum reasonably foreseeable accident because it could (1) cause a major breach of confinement barriers, (2) involve a large portion of the material at risk, and (3) have a high-energy release mechanism (physical impact followed by a sustained fire). The analysis eliminated other accident scenarios considered in this frequency range because they would not have sufficient energy sources to cause a large breach of confinement and release to the atmosphere. Although the facility contains little combustible material to sustain a fire, a fire caused by aircraft fuel involved in the crash could increase potential consequences over other beyond-design-basis accidents. The major events of an aircraft crash scenario are as follows:

- A large or high-velocity aircraft (e.g., commercial or military) crashes directly into the Hot Fuel Examination Facility.

- The impact has sufficient force to cause catastrophic failure of the building structure, breach of the Main Cell, and loss of forced cooling to subassemblies in the cell.

- The fuel in the aircraft is released to the facility and is ignited.

- The ensuing fire involves the contents of the Main Cell, Decontamination Cell, High Bay Area, and Hot Repair Area, resulting in atmospheric release of radionuclides.

To determine aircraft crash probability, the analysis limited this scenario to large or high-velocity jet airplanes. High-velocity military jets from the U.S. Air Force Base at Mountain Home in southwestern Idaho could enter the airspace of the INEL. In addition, large jet aircraft have been flown at low altitudes in landing configurations over portions of the INEL for vortex tests. The likelihood of a large aircraft crash directly in the Hot Fuel Examination Facility is remote, but possible. Analyses of jet aircraft crashes at specific facilities, such as the Idaho Chemical Processing Plant, have resulted in predicted frequencies on the order of 1.0×10^{-7} event per year. Because specific analyses have not determined the likelihood of an aircraft crash into the Hot Fuel Examination Facility (although it is expected that fewer flights occur over the Argonne National Laboratory-West facility area than the Idaho Chemical Processing Plant), the analysis conservatively assumed that the frequency for an aircraft crashing into the Hot Fuel Examination Facility is 1.0×10^{-7} per year.

For determining impacts from this postulated accident scenario, the analysis assumed the worker was located 100 meters from the event; and the nearest point of public access (U.S. Route 20) and the nearest site boundary were both at 5,240 meters (5,730 yards).

5.15.6.5 Accident 5: Inadvertent Nuclear Chain Reaction During Spent Nuclear Fuel Processing (1 x 10^{19} fissions) at the Idaho Chemical Processing Plant CPP-666 Fluorinel and Storage (FAST) Facility. The accident screening methodology discussed in Section 5.15.3 identified an inadvertent nuclear criticality resulting from spent nuclear fuel reprocessing in the CPP-666 Fluorinel and Storage Facility as a maximum reasonably foreseeable processing accident. Although the CPP-666 Fluorinel and Storage Facility, which historically reprocessed spent nuclear fuel to recover fissionable radionuclides (e.g., uranium-235), is currently shutdown, there may be a need to resume processing operations to dissolve spent nuclear fuel and to stabilize the radionuclides in a waste form. Therefore, while the potential for this accident does not currently exist, the potential would exist if processing-related activities are resumed under Alternatives 4b(1) and 5b (Regionalization and Centralization at the INEL, respectively).

Initiating events that the analysis considered possible to lead to an inadvertent nuclear criticality during processing included human error, equipment failure, an earthquake, an aircraft crash, excessive fissionable radionuclides in the spent nuclear fuel being processed, and reduced neutron poison concentrations. Consistent with the inadvertent criticality scenario associated with underwater storage of spent nuclear fuel described in Section 5.15.6.2, the fission yield associated with this criticality was assumed to be 1×10^{19} fissions. Further information and references regarding this postulated accident scenario are provided in Slaughterbeck et al. (1995) and EG&G (1993b).

As discussed in Section 5.15.2, three inadvertent nuclear criticalities have occurred in INEL processing facilities during the 40-year history of the INEL. The last of these criticalities occurred 14 years ago. As a result of these accidents, administrative controls and facility modifications were implemented to reduce the potential for inadvertent nuclear criticality accidents resulting from processing-related activities. If the decision is made to resume processing operations, these same controls would be utilized. Therefore, the estimated frequency for a potential inadvertent nuclear criticality is assumed to be 1×10^{-3} events per year, which is consistent with assumptions made regarding the potential for an inadvertent criticality resulting from underwater storage and handling of severely degraded spent nuclear fuel (as discussed in Section 5.15.6.2).

Limited credit was taken for mitigative features, such as emergency response programs, in determining worker and public exposures resulting from this postulated accident scenario. However, credit was taken for shielding walls placed in the facility to reduce potential personnel exposures resulting from an inadvertent nuclear criticality.

To determine the accident impacts from this postulated accident scenario, the analysis assumed the worker to be located 100 meters (328 feet) from the event, the nearest point of public access (U.S., Route 20/26) is 5,870 meters (6,420 yards), and the nearest site boundary is located at 14,000 meters (15,310 yards).

5.15.6.6 Accident 6: Radionuclide Release During Spent Nuclear Fuel Processing at the Idaho Chemical Processing Plant CPP-666 Fluorinel and Storage (FAST) Facility Resulting from a Hydrogen Explosion in the Dissolver Off-Gas System. The accident screening methodology discussed in Section 5.15.3 identified a hydrogen explosion in the CPP-666 Fluorinel and Storage Facility dissolver off-gas system as a maximum reasonably foreseeable processing accident. Despite CPP-666's current shutdown status, there may be a need to resume processing operation to dissolve spent nuclear fuel and stabilize the radionuclides in a waste form. Therefore, while the potential for this accident does not currently exist, the potential would exist if processing-related activities are resumed under Alternatives 4b(1) and 5b (Regionalization and Centralization at the INEL, respectively).

Initiating events that the analysis considered possible to lead to a hydrogen explosion in the dissolver off-gas system included human error, equipment failure, and an earthquake. Further information and references regarding this postulated accident scenario are provided in Slaughterbeck et al. (1995) and EG&G (1993b).

Limited credit was taken for mitigative features, such as emergency response programs, in determining worker and public exposures resulting from this postulated accident scenario. To determine the accident impacts from this postulated accident scenario, the analysis assumed the worker to be located 100 meters (328 feet) from the event, the nearest point of public access (U.S., Route 20/26) is 5,870 meters (6,420 yards), and the nearest site boundary is located at 14,000 meters (15,310 yards).

5.15.6.7 Accident 7: Radionuclide Release During Spent Nuclear Fuel Processing at the Idaho Chemical Processing Plant CPP-666 Fluorinel and Storage (FAST) Facility Resulting from the Inadvertent Dissolution of 30-Day Cooled Spent Nuclear Fuel. The accident screening methodology discussed in Section 5.15.3 identified a radionuclide release resulting from the inadvertent dissolution of 30-day cooled spent nuclear fuel in the CPP-666 Fluorinel and Storage Facility as a maximum reasonably foreseeable accident. There may be a need to resume processing operation at CPP-666 to dissolve spent nuclear fuel and stabilize the radionuclides in a waste form. Therefore, while the potential for this accident does not currently exist, the potential would exist if processing-related activities are resumed under Alternatives 4b(1) and 5b (Regionalization and Centralization at the INEL, respectively).

Upon removal from a nuclear reactor, spent nuclear fuel is placed in an underwater storage canal (e.g., Advanced Test Reactor Storage Canal in the Test Reactor Area) to allow the fuel temperature to cool and short-lived radionuclides to decay. Inadvertent processing of spent nuclear fuel that has not had the opportunity to sufficiently cool presents the potential for accidents during dissolution of the fuel. Examples of accidents that could potentially occur are explosions in the dissolver tank and an inadvertent criticality. An explosion resulting from inadvertent dissolving spent nuclear fuel that has not sufficiently cooled (i.e., 30-day cooled fuel) is considered for this analysis since an inadvertent criticality is already considered (as discussed in Section 5.15.6.6).

The potential initiating event considered for this accident involves several operator errors that result in the wrong spent nuclear fuel assemblies being dissolved. First, fuel cooled 30 or fewer days would have to be shipped to and received by the Fluorinel and Storage Facility. Second, operators at the CPP-666 Fluorinel and Storage Facility would have to inadvertently dissolve the 30-day (or fewer) cooled fuel. Based on the individual probability of these events, and the probability that the dissolved fuel would accidentally release radionuclides to the environment, the estimated frequency for this event is 1×10^{-6} events per year. Further information and references regarding this postulated accident scenario are provided in Slaughterbeck et al. (1995) and EG&G (1993b).

Limited credit was taken for mitigative features, such as emergency response programs, in determining worker and public exposures resulting from this postulated accident scenario. To determine the accident impacts from this postulated accident scenario, the analysis assumed the worker to be located 100 meters (328 feet) from the event, the nearest point of public access (U.S., Route 20/26) is 5,870 meters (6,420 yards), and the nearest site boundary is located at 14,000 meters (15,310 yards).

5.16 Cumulative Impacts and Impacts from Connected or Similar Actions

The INEL already contains major DOE facilities unrelated to spent nuclear fuel that would continue to operate throughout the life of the spent nuclear fuel management program. The activities associated with these existing facilities produce environmental consequences that this EIS has included in the baseline environmental conditions (Chapter 4) against which it has assessed the consequences of the spent nuclear fuel alternatives. In addition, the cumulative impacts assessed in this section include other past, present, and reasonably foreseeable future actions that DOE expects to occur at the INEL, such as spent nuclear fuel management, Naval Nuclear Propulsion Program activities, environmental restoration and waste management activities, as well as any known offsite projects conducted by government agencies, businesses, or individuals. Onsite projects include decontamination and decommissioning, repair, and upgrades of existing facilities. Offsite projects include residential and commercial development, and changes in manufacturing plants.

Consistent with the DOE sliding scale approach and the programmatic aspects of this EIS, cumulative impacts are discussed commensurate with the degree of impact. Therefore, not every area of analysis from Chapter 5 is represented in this section. DOE used information and analyses from Volume 2 of this EIS as input for this section. Section 5.15 of Volume 2 provides a more detailed discussion of cumulative impacts.

Tables 5.16-1 and 5.16-2 list the cumulative impacts identified for each alternative. DOE made necessary adjustments to accommodate the differences between Volume 1 and Volume 2 alternatives. Cumulative impacts from Alternatives 3 and 4a are nominally the same, as are cumulative impacts from Alternatives 1 and 2, 5a and 4b(2), and 5b and 4b(1).

5.16.1 Land Use

Implementation of any of the alternatives would contribute to the cumulative loss of land with open-space land use. However, the cumulative amount of land that would no longer be open space or available for other land uses would be small compared to the size of INEL or regional land uses. As discussed in Section 5.2, Land Use, the maximum land disturbance, 31 acres (0.12 square kilometer) would occur under Alternative 4b(l) [Regionalization by Geography (INEL)] and 5b (Centralization at INEL). While exact maximum figures are not available, over 200 acres (0.81 square kilometer) of vacant land in nearby communities are scheduled for development. Projects that would potentially

Table 5.16-1. Nonhealth-related cumulative impacts.

Discipline/Unit of measure		1 (No Action) and 2 (Decentralization)	3 (1992/1993 Planning Basis) and 4a (Regionalization by Fuel Type)	5a (Centralization at Other Sites) and 4b(2) [Regionalization by Geography (Elsewhere)]	5b (Centralization at INEL) and 4b(1) [Regionalization by Geography (INEL)]	Comments
Land use/amount of land not available for other use		Small compared to regional land uses	Small compared to regional land uses	Small compared to regional land uses	Small compared to regional land uses	
Socioeconomics/change in number of total jobs		Overall decrease of 4,800	Overall decrease of 2,300	Overall decrease of 4,400	Overall decrease of 1,400	Under all alternatives, additional jobs created would be more than offset by decrease from other actions
Cultural resources/minimum number of potentially historic structures/archaeological sites disturbed[a]		6 structures and 0 sites	70 structures and 22 sites	11 structures and 0 sites	70 structures and 22 sites	Under all alternatives, the potential for reduction of the number of cultural resources exists
Air resources[b]		Below applicable standards	Below applicable standards	Below applicable standards	Below applicable standards	
Waste management/waste volume total pending disposition	High-level[d]	12,100 m^3	12,500 m^3	17,000 m^3	12,100 m^3	These volumes reflect existing and newly generated wastes pending disposition under each alternative
	Transuranic[c]	67,000 m^3	73,000 m^3	67,000 m^3	87,000m^3	
	Mixed low-level	17,000 m^3	17,000 m^3	17,000 m^3	167,000 m^3	
	Low-level[e]	46,000 m^3	72,000 m^3	47,000 m^3	840,000 m^3	
	Hazardous[f]	12,000 m^3	12,000 m^3	12,000 m^3	12,000 m^3	
	Commercial and industrial[e]	540,000 m^3	590,000 m^3	550,000 m^3	590,000 m^3	

a. Numbers for archaeological sites potentially impacted would be expected to increase as cultural resource surveys are conducted for projects on acreage previously unsurveyed.
b. See Table 5.16-2 for cumulative health risks related to air emissions.
c. Derived in Freund (1994), Morton and Hendrickson (1995).
d. High-level waste includes both liquid and calcine forms. Liquid high-level waste totals do not include processing, which would increase these reported totals by some degree. Numbers represent total volume of all high-level waste stored onsite.
e. Numbers do not include existing dispositioned waste stored or buried onsite.
f. Numbers represent total volume stored onsite.

Table 5.16-2. Health-related cumulative impacts.

Radiological[a]	Pathway	Type of impact	1 (No Action) and 2 (Decentralization)	3 (1992/1993 Planning Basis) and 4a (Regionalization by Fuel Type)	5a (Centralization at Other Sites) and 4b(2) [Regionalization by Geography (Elsewhere)]	5b (Centralization at INEL) and 4b(1) [Regionalization by Geography (INEL)]	Comments
Public	Atmoshperic	Estimated excess fatal cancers	<1	<1	<1	<1	
	Groundwater	Estimated excess fatal cancers	<1	<1	<1	<1	
	Biotic	Estimated excess fatal cancers	<1	<1	<1	<1	This pathway would involve harvesting game animals and vegetation that can assimilate radioactivity onsite.
Workers[b]	Atmospheric	Estimated excess fatal cancers	Negligible	Negligible	Negligible	Negligible	Overall cancers expected to be less than baseline because fewer employees under all alternatives.
	Occupational exposures	Estimated excess fatal cancers	1	1	1	1	
Public	Atmospheric (Carcinogens)	Estimated lifetime cancers	<1	<1	<1	<1	
	Atmospheric (Noncarcinogens)[c]	Estimated adverse health effects	0	0	0	0	

Table 5.16-2. (continued).

Radiological[a]	Pathway	Type of impact	1 (No Action) and 2 (Decentralization)	3 (1992/1993 Planning Basis) and 4a (Regionalization by Fuel Type)	5a (Centralization at Other Sites) and 4b(2) [Regionalization by Geography (Elsewhere)]	5b (Centralization at INEL) and 4b(1) [Regionalization by Geography (INEL)]	Comments
Workers[b]	Atmospheric (Carcinogens)	Estimated lifetime cancers	<1	<1	<1	<1	
	Atmospheric (Noncarcinogens)[c]	Estimated adverse health effects	0	0	0	0	
	Routine workplace safety hazards	Estimated fatalities	3	3	3	3	Estimates differ only slightly between alternatives due to changes in number of workers. Total workplace safety hazards are fewer than those encountered by the average worker in private industry.

a. Approximate numbers. See Volume 2, Section 5.12 and Volume 2, Appendix F for detailed discussion and analyses.
b. Estimated excess fatal cancers calculated from dosimeter measurements.

disturb previously disturbed land are scheduled to take place on about 270 acres (1.0 square kilometer) at the INEL. An additional 1,060 acres (4.3 square kilometers) of open space INEL land may also be disturbed by potential projects.

5.16.2 Socioeconomics

Any of the spent fuel management alternatives would cause minimal cumulative impacts on socioeconomic resources of the INEL region when combined with known onsite or offsite projects. The implementation of any of the alternatives would create temporary additional employment during construction; the upper bound of potential impact would occur under Alternatives 3, 4a, 4b(1), and 5b. In the long term, the expected future decrease in employment at the INEL would more than offset this increase, as well as any increases from known offsite projects. Therefore, the cumulative effect on employment would be an overall decrease. Potential population declines associated with the cumulative effect on regional employment are estimated to represent less than 2 percent of the total regional population. It is unlikely that a change in population of this size would generate any notable long-term adverse impacts to housing, community services, or public finance in the region.

5.16.3 Cultural Resources

The types of cumulative impacts on cultural resources are the same for all alternatives. Each of the alternatives, when combined with associated onsite and offsite activities, could potentially impact cultural resources. However, surveying, recording, and stabilizing archeological and historic sites and structures at the INEL would increase scientific knowledge of the region's cultural resources, although stabilizing resources may adversely affect their significance to Native American groups. The unchecked deterioration of both structures and historic documents on nuclear facilities at the INEL could have a long-term adverse impact on these resources. Long-term effects may also occur to traditional resources that may not be mitigated through scientific studies. Cumulative impacts associated with Alternatives 3 and 4a (see 1992/I993 Planning Basis and Regionalization by Fuel Type) and Alternatives 5b and 4b(1) [Centralization at INEL and Regionalization by Geography (INEL)] have the greatest potential for impacts. Alternatives 1 and 2 (No Action and Decentralization) would have the least potential for impacts.

5.16.4 Air Quality

For radiological emissions, all cumulative impacts at onsite and offsite locations are well below applicable standards and are a small fraction of the dose received from natural background sources. The highest dose to a maximally exposed member of the public would be caused by Alternatives 4b(l) and 5b and would be about 0.05 millirem per year. When added to the projected dose from other INEL proposed projects of approximately 0.7 millirem per year and the maximum baseline dose of 0.05 millirem per year, this dose would be well below the National Emissions Standards for Hazardous Air Pollutants limit of I0 millirem per year (CFR 1992c). The National Council on Radiation Protection and Measurements has identified a dose rate below 1 millirem per year as negligible (NCRP 1987).

Cumulative nonradiological impacts were analyzed in terms of concentrations of criteria and toxic air pollutants in ambient air. At site boundary locations, the highest potential concentrations of criteria pollutants remain well below applicable National Ambient Air Quality Standards (CFR 1991). Concentrations at public road locations within the INEL boundary could increase significantly from current levels, but would remain well below applicable standards.

5.16.5 Occupational and Public Health and Safety

Work activities and the exposure to radiological and chemical hazards under each of the alternatives would be similar to those at present. Therefore, average radiation dose, exposure to toxic chemicals, and associated health effects would be related to the number of site workers under each alternative. Because the cumulative impacts of any alternative would be a decrease in the number of workers, the cumulative impact of any alternative on occupational health would be a decrease in health effects to the levels listed in Table 5.16-2. The incidence of expected health effects would be similar for all alternatives because the relative difference in employment effects (and therefore the effects on the health of those employed) is very small. While air emissions present the only calculable pathway for public radiation exposure due to spent nuclear fuel management, groundwater and biotic pathways are included in Table 5.16-2 due to Volume 2 analyses of environmental restoration and waste management activities.

Occupational health data concerning historic accidents are incomplete and not readily available. Though historical records of accidents at the INEL are available, occupational doses were not always known and reported. Worker dose data are currently being collected and analyzed under a National

Institute of Occupational Safety and Health program. Historical offsite doses associated with the INEL are summarized in the Idaho National Engineering Laboratory Historical Dose Evaluation (DOE 1991). The Centers for Disease Control and Prevention is conducting a more comprehensive reconstruction of doses from INEL operations. An assessment of the cumulative impacts of accidents at the Site to the health of INEL workers is not available at this time.

Cumulative transportation impacts are addressed in Volume 1, Appendix I.

5.16.6 Materials and Waste Management

The total volumes of waste existing and projected to be generated or shipped to the INEL from spent nuclear fuel management, as well as known onsite and offsite projects over a 10-year period, are presented by waste stream for each alternative in Table 5.16-1. The storage of low-level waste for incineration is not considered to be restrictive between 1995 and 2005; however, beyond 2005 additional capacity may be required. Although spent nuclear fuel management would not cause permitted storage capacity to exceed its limits without available treatment or disposal under the No Action and Decentralization Alternatives, it is anticipated that the permitted storage capacity for mixed low-level waste will be exceeded during the first year of a 10-year timeframe. All other alternatives include facility construction for storage of, or shipping of, mixed low-level waste; therefore, storage capacity is accounted for.

5.17 Adverse Environmental Effects That Cannot Be Avoided

The construction and operation of any of the alternatives at the INEL could result in adverse impacts to the environment. Changes in project design and other measures would avoid or otherwise mitigate most of these impacts to minimal levels. This section identifies only adverse impacts that mitigation could not reduce to minimal levels or avoid altogether.

Under each alternative, the continued deterioration of structures with historic preservation potential and historic documents on nuclear facilities could have a long-term adverse impact on these resources at the INEL. However, DOE would avoid potentially adverse impacts by preserving the historic value of the property through appropriate research, or by conducting limited rehabilitation on these structures. This impact is discussed in Section 5.4.

As discussed in Section 5.2, the maximum loss of habitat would involve the conversion to industrial use of about 31 acres (0.12 square kilometers) of previously disturbed habitat that is of low quality and limited use to wildlife; conversion would occur under Alternatives 4b(1) and 5b.

The amount of radiation exposure from normal operation of the spent nuclear fuel facilities would be a small fraction of the existing natural background at the INEL and would be well below applicable regulatory standards. In all cases, the number of estimated additional cancers is a small fraction of 1 per year of site operation through 2035. This effect is discussed in Section 5.12.

With the exception of the unavoidable temporary increase in noise due to construction activities, any impact of noise from activities under any of the alternatives would be minor and highly unlikely.

An unavoidable adverse impact of the proposed activities with any of the alternatives would be an accident either at the involved facilities or during the transportation of construction materials or dismantled components. Accidents are discussed in Section 5.15; transportation is discussed in Section 5.11.

Spent nuclear fuel management supports the continuation of beneficial activities such as radiopharmaceutical and other research. An unavoidable adverse impact of the No-Action Alternative would be a reduction in the support of such activities.

As discussed in Section 5.14, the increased generation of industrial solid waste that would occur under all alternatives is an unavoidable adverse impact. However, the amount generated under each alternative would be a very small percentage increase from the projected 1995 baseline levels.

5.18 Relationship Between Short-Term Use of the Environment and the Maintenance and Enhancement of Long-Term Productivity

Under all alternatives, short-term use of the environment is generally associated with resource demands for spent nuclear fuel management activities. Resources demands also include those required for upgrade, construction, and operation of facilities. These short-term demands and uses provide a foundation and direction for the long-term productivity of INEL; they also have an effect on the success of future INEL missions. A brief discussion of the influence proposed actions would have on the long-term productivity of the INEL follows. The INEL missions, including spent nuclear fuel, are discussed in Section 2.1.

The No-Action Alternative would provide few long-term benefits and would not allow DOE-Idaho Operations Office to fulfill its missions regarding the disposition and management of spent nuclear fuel. The activities proposed in this alternative would not support future proposals for disposal technology development. Further, the No-Action Alternative could bring enforcement actions because it would not meet all the requirements of existing DOE regulatory commitments such as those outlined in the Federal Facility Agreement and Consent Order.

To a varying degree, Alternatives 2, 3, and 4(a) would provide more flexibility than other alternatives for fulfilling existing or future missions and actions at INEL. Near- and long-term actions under these alternatives ensure compliance with regulatory requirements and protection of the environment. Furthermore, these alternatives would provide a diverse decisionmaking platform for future actions concerning disposition of DOE spent nuclear fuel. Facilities constructed and technologies developed under these alternatives could be used for a wide range of activities such as interim treatment and storage or preparation and packaging for transportation offsite.

The approach that would be taken for spent nuclear fuel under Alternatives 4b(2) and 5a could confine and hinder long-term productivity at INEL. Efforts would focus on shipment of spent nuclear fuel to other locations. No emphasis would be placed on solving particular spent nuclear fuel disposal problems or increasing the understanding of how certain spent nuclear fuels react over time.

Alternatives 4b(1) and 5b would direct INEL's future mission and development primarily toward large-scale canning and characterization, storage, and disposal of all INEL and DOE regional or complex-wide spent nuclear fuel. These alternatives could limit INEL's flexibility in redirecting or enhancing future INEL-specific missions.

5.19 Irreversible and Irretrievable Commitment of Resources

The irreversible and irretrievable commitment of natural and manmade resources resulting from the construction and operation of facilities related to the spent nuclear fuel alternatives would involve materials and resources that could not be recovered or recycled or that would be consumed or reduced to unrecoverable forms. Some of these commitments would be irretrievable because of the nature of the commitment or the cost of reclamation. For example, the construction and operation of spent nuclear fuel facilities at the INEL would consume irretrievable amounts of electrical energy, fuel, concrete, steel, aluminum, copper, plastics, lumber, sand, gravel, groundwater, and miscellaneous chemicals.

Alternatives 4b(1) and 5b are each estimated to require approximately 11,000 megawatt-hours per year of electricity, 1,100,000 liters (290,000 gallons) per year of fuel oil, and 48 million liters (13 million gallons) per year of water above the projected baseline (1995) usage of these resources (see Section 5.13). These changes would represent a modest increase of 5.3 percent, 9.9 percent, and 0.7 percent respectively, and are well within current system capabilities and usage limits. All other alternatives would place smaller demands on these resources, commensurate with the level of construction and operation activities proposed.

Alternatives 4b(1) and 5b would also commit 31 acres (0.12 square kilometer) of previously disturbed land to industrial use; the conversion of this acreage would result in the commitment of poor quality wildlife habitat and natural resource services. Alternatives 4b(1) and 5b would involve the greatest irretrievable consumption of other resources, such as construction materials and operating supplies. However, this demand would not constitute a permanent drain on local resources or involve any material that is in short supply in the region.

Other commitments would be irreversible because the construction or operation of facilities related to the spent nuclear fuel alternatives would consume the resource. Proposed activities would also require an expenditure of labor that would be irretrievable.

5.20 Potential Mitigation Measures

This section summarizes measures that DOE would use to avoid or reduce impacts to the environment caused by spent nuclear fuel management activities at the INEL. The potential mitigation measures for each aspect of the affected environment described below are the same under each alternative. Section 5.7 of Volume 1 discusses other generalized measures DOE could use.

5.20.1 Pollution Prevention

DOE is committed to comply with Executive Order 12856, Federal Compliance with Right-to-Know Laws and Pollution Prevention Requirements; Executive Order 12873, Federal Acquisition, Recycling and Waste Prevention; and applicable DOE Orders and guidance documents in planning and implementing pollution prevention at the INEL. The DOE views source reduction as the first priority in its pollution prevention program, followed by an increased emphasis on recycling. Waste treatment and disposal are considered only when prevention or recycling is not possible or practical.

5.20.2 Cultural Resources

The lack of detailed specifications associated with the proposed construction at the INEL under various alternatives precludes identifying specific project impacts and potential mitigation measures for particular structures and facilities. Basic compliance under cultural resource law involves five steps that would be essentially the same under all alternatives. These steps are (a) identification and evaluation of resources in danger of impact, (b) assessment of effects to these resources in consultation with the State Historic Preservation Office and representatives of the Shoshone-Bannock Tribes, (c) development of plans and documents to minimize any adverse effects, (d) consultation with the Advisory Council on Historic Preservation and tribal representatives as to the appropriateness of mitigation measures, and (e) implementation of potential mitigation measures. Therefore, if a cultural resource survey has not been performed in an area planned for ground disturbance under one of the proposed alternatives, consultation would be initiated with the Idaho State Historic Preservation Office and the survey would be conducted prior to any disturbance. If cultural resources were discovered, they would be evaluated according to National Register criteria. Wherever possible, important resources would be left undisturbed. If the impacts are determined to be adverse and it is not feasible to leave the resource undisturbed, then measures would be initiated to reduce impacts. All mitigation plans would be developed in consultation with the State Historic Preservation Office and the Advisory Council on Historic Preservation and would conform to appropriate standards and guidelines established for historic preservation activities by the Secretary of the Interior.

Some actions may affect areas of religious, cultural, or historic value to Native Americans. DOE has implemented a Working Agreement (DOE 1992d) to ensure communication with the Shoshone-Bannock Tribe, especially relating to the treatment of archeological sites during excavation, as mandated by the Archeological Resources Protection Act (ARPA 1979); the protection of human remains, as required under the Native American Graves Protection and Repatriation Act (NAGPRA 1990); and the free exercise of religion as protected by the American Indian Religious Freedom Act (AIRFA 1978). In keeping with DOE Native American policy (DOE 1990), DOE Order 1230.2 (DOE 1992c), and procedures to be defined in the final Cultural Resources Management Plan for the INEL, DOE would conduct Native American consultation during the planning and implementation of all proposed alternatives. Procedures for dealing with the inadvertent discovery of human remains would be consistent with the Native American Graves Protection and Repatriation Act (NAGPRA 1990). If human remains are discovered, DOE will notify all tribes that have expressed an interest in the repatriation of graves as required under NAGPRA, including the Shoshone-Bannock, Shoshone, Painte, and the Northwestern band of the Shoshone Nation. These tribes will then have an opportunity to claim the remains and associated artifacts in accordance with the requirements of NAGPRA. Procedures for the repatriation of "cultural items" in accordance with NAGPRA will be described in a curation agreement that will be finalized by June 1996.

In addition to consultation, other measures would mitigate potential adverse effects to Native American Resources, in particular effects to air, water, plants, animals, and visual setting. These measures include avoidance of sensitive areas, placement of facilities within existing areas of construction, revegetation with native plants of areas with ground disturbance, monitoring of plants and animals within hunting and gathering areas for radiological contamination, reducing noise and night lights outside of existing facilities, monitoring tanks, ponds and runoff for contaminants, minimizing ground disturbance, use of dust suppressers during construction, and use of filters and other air pollutant control equipment to reduce air contaminants.

5.20.3 Traffic and Transportation

All onsite shipments of spent nuclear fuel would be in compliance with ID Directive 5480.3, "Hazardous Materials Packaging and Transportation Safety Requirements." These requirements provide assurance that, under normal conditions, the INEL would meet as-low-as-reasonably-achievable conditions, reasonably foreseeable accident situations (those with probability of occurrence greater than 1×10^{-7} per year) would not result in a loss of shielding or containment or a criticality, and an unintentional release of radioactive material would result in a timely response.

DOE would approve the type packages used for onsite shipments or would obtain a Nuclear Regulatory Commission or DOE certificate of compliance. If the onsite package did not have Nuclear Regulatory Commission or DOE certification, the user of the package would have to establish how administrative controls or other potential mitigating measures would ensure that the package would maintain containment and shielding integrity. The administrative and emergency response considerations would provide sufficient control so that accidents would not result in loss of containment or shielding, in criticality, or in an uncontrolled release of radioactive material that would create a hazard to the health and safety of the public or workers. Accident mitigation is described below.

5.20.4 Accidents

The DOE would initiate INEL emergency response programs, as appropriate, following the occurrence of an accident to prevent or mitigate consequences. These emergency response programs, implemented in accordance with 5500-DOE series Orders, typically involve emergency planning, emergency preparedness, and emergency response actions. Participating government agencies with plans that are interrelated with the INEL Emergency Plan for Action include the State of Idaho, Bingham County, Bonneville County, Butte County, Clark County, Jefferson County, the Bureau of Indian Affairs, and Fort Hall Indian Reservation. When an emergency condition exists at a facility, the Emergency Action Director is responsible for recognition, classification, notification, and protective action recommendations. Each emergency response plan utilizes resources specifically dedicated to assist a facility in emergency management. These resources include but are not limited to the following:

- INEL Warning Communications Center
- INEL Fire Department
- Facility Emergency Command Centers
- DOE Emergency Operations Centers
- County and State Emergency Command Centers
- Medical, health physics, and industrial hygiene specialists

- Protective clothing and equipment (respirators, breathing air supplies, etc.)
- Periodic training exercises and drills within and between the organizations involved in implementing the response plans

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Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement

Volume 1
Appendix C

Savannah River Site
Spent Nuclear Fuel Management Program

April 1995

U.S. Department of Energy
Office of Environmental Management
Idaho Operations Office

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FIGURES

1. INTRODUCTION

The U.S. Department of Energy (DOE) is engaged in two related decisionmaking processes concerning: (1) the transportation, receipt, processing, and storage of spent nuclear fuel (SNF) at the DOE Idaho National Engineering Laboratory (INEL) which will focus on the next 10 years; and (2) programmatic decisions on future spent nuclear fuel management which will emphasize the next 40 years.

DOE is analyzing the environmental consequences of these spent nuclear fuel management actions in this two-volume Environmental Impact Statement (EIS). Volume 1 supports broad programmatic decisions that will have applicability across the DOE complex and describes in detail the purpose and need for this DOE action. Volume 2 is specific to actions at the INEL. This document, which limits its discussion to the Savannah River Site (SRS) spent nuclear fuel management program, supports Volume 1 of the EIS. Other documents supporting Volume 1 focus on spent nuclear fuel management programs for the Hanford Site, INEL, Naval Nuclear Propulsion Program, and other sites.

As part of its planning process for this two-volume EIS, DOE issued an Implementation Plan on October 29, 1993. The organization of this document is consistent with the provisions established in the Implementation Plan and are outlined below:

- Chapter 2 contains background information related to the SRS and the framework of environmental regulations pertinent to spent nuclear fuel management.

- Chapter 3 identifies spent nuclear fuel management alternatives that DOE could implement at the SRS, and summarizes their potential environmental consequences.

- Chapter 4 describes the existing environmental resources of the SRS that spent nuclear fuel activities could affect.

- Chapter 5 analyzes in detail the environmental consequences of each spent nuclear fuel management alternative and describes cumulative impacts. The chapter also contains information on unavoidable adverse impacts, commitment of resources, short-term use of the environment and mitigation measures.

2. BACKGROUND

The chapter contains an overview of the Savannah River Site (SRS) and a description of the regulatory framework related to the actions that this document evaluates. In addition, it discusses the U.S. Department of Energy (DOE) Spent Nuclear Fuel (SNF) Management Program as it relates to the SRS. Finally, it describes the representative sites located on the SRS that could serve as locations for spent nuclear fuel facilities.

2.1 SRS Overview

The SRS is a key DOE facility for research on and processing of special nuclear materials. The U.S. Government built the Site in the early 1950s to produce the basic materials - primarily plutonium-239 and tritium - used in the fabrication of nuclear weapons. The DOE Savannah River Operations Office manages the SRS, and Westinghouse Savannah River Company (WSRC) operates the Site under contract to DOE.

2.1.1 Site Description

The SRS occupies an area of approximately 310 square miles (800 square kilometers) in western South Carolina, in a generally rural area about 25 miles (40 kilometers) southeast of Augusta, Georgia, and 12 miles (19 kilometers) south of Aiken, South Carolina (Figure 2-1). The Savannah River forms the southwestern border of the SRS, which includes portions of Aiken, Barnwell, and Allendale Counties. The average population density (1990 census data) in the six-county region of influence around the Site is 140 people per square mile (54 per square kilometer); the largest concentration is 2,595 people per square mile (1,002 per square kilometer) in the City of Augusta (HNUS 1992). Four other population centers — Aiken, Allendale, Barnwell, and North Augusta, South Carolina — are within 22 miles (40 kilometers) of the Site. Three small towns — Jackson, New Ellenton, and Snelling, South Carolina — are adjacent to the SRS boundary to the northwest, north, and east, respectively. Based on 1990 U.S. Census Bureau data, the population within a 50-mile (80-kilometer) radius of the SRS is approximately 620,100 (Arnett et al. 1993).

The Site consists primarily of managed upland forest with some wetland areas. Facilities and roadways occupy approximately 5 percent of the SRS land area. Access to the Site is controlled, with

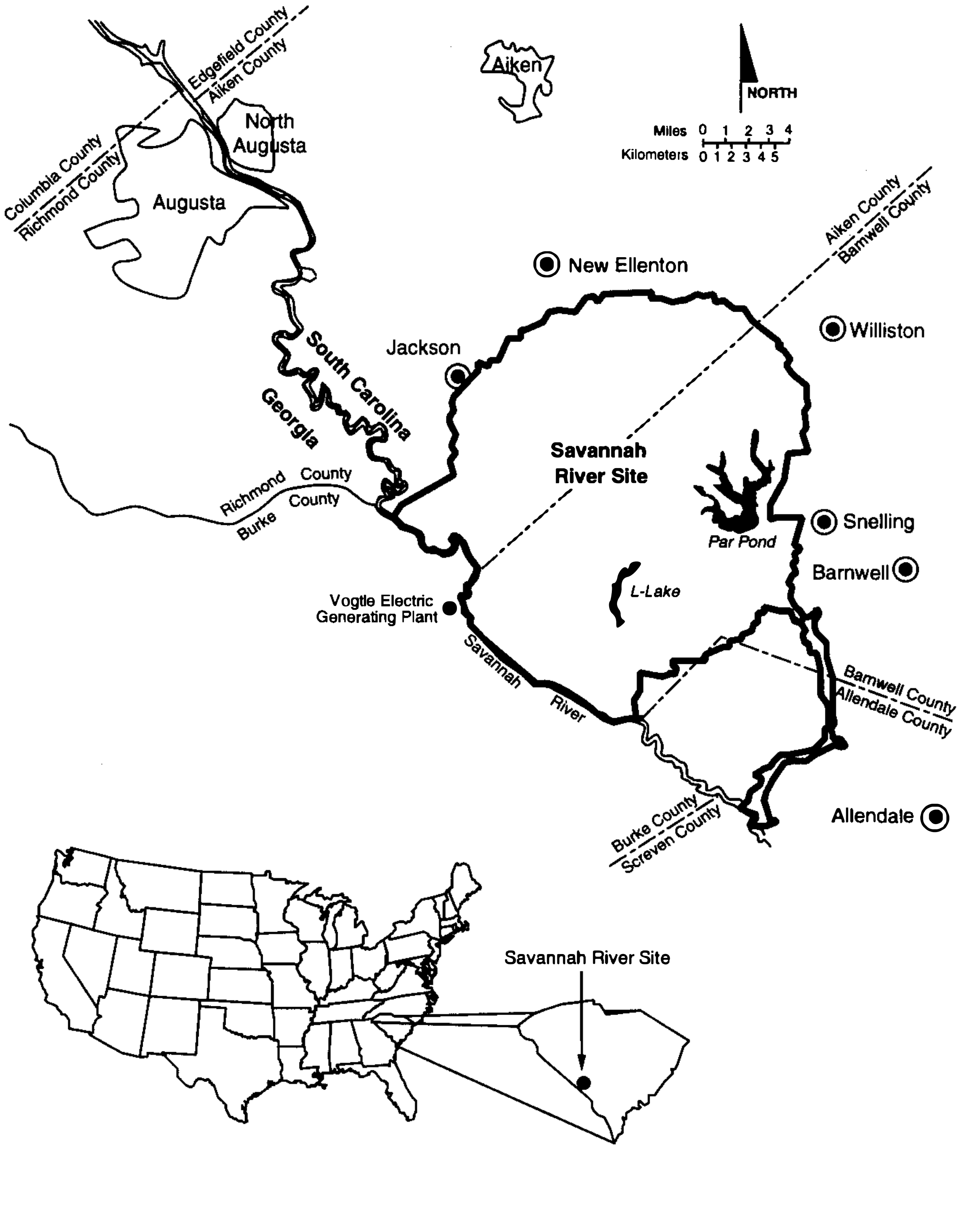

Figure 2-1. National location of SRS.

public transportation limited to through traffic on South Carolina Highway 125 (SRS Road A), U.S. Highway 278, SRS Road 1, and the CSX Railroad corridor.

The SRS contains 15 major production, service, and research and development (R&D) areas that previously supported nuclear materials production and can support processing operations and waste management activities. Major SRS facilities include five nuclear reactors, two chemical separations plants, a fuel and target fabrication facility, the Defense Waste Processing Facility (DWPF), the Replacement Tritium Facility, a heavy-water rework plant, and the Savannah River Technology Center (SRTC), formerly called the Savannah River Laboratory. In addition, the University of Georgia Research Foundation operates the Savannah River Ecology Laboratory (SREL) on the Site under contract to DOE. Under an interagency agreement, the U.S. Forest Service operates the Savannah River Forest Station, which manages the natural resources and secondary roads on the Site. These facilities are in defined areas scattered across the Site. Each area is identified by a letter designation, as summarized in Table 2-1. Figure 2-2 shows the locations of the principal SRS facilities. The reactor, waste storage, and separations areas are at least 4 miles (6 kilometers) inside the nearest SRS boundary.

The primary SRS facilities were related to the production of nuclear materials. M-Area manufactured fuel and target components for shipment to the SRS reactors. Originally, the Site operated five reactors; at present, all are in shutdown status. Shielded railroad cars transported irradiated fuel to the F- or H-Area Canyon for the recovery of nuclear materials. The F- and H-Area separations processes dissolve irradiated components in acid, and extract and separate the desired nuclear materials. In H-Area, additional processes extract other products from irradiated components.

DOE neutralizes and stores the high-level liquid radioactive waste generated by the separations facilities in underground tanks. DOE plans to process this waste into a borosilicate glass waste form in the Defense Waste Processing Facility when that facility becomes operational, and to store this glass waste form at the SRS until an offsite geological repository is available. [DOE has prepared a Supplemental EIS related to Defense Waste Processing Facility operations (DOE 1994a).] In addition to the underground waste storage tanks, DOE has established a centrally located 196-acre (0.8-square-kilometer) site between F- and H-Areas, called E-Area, for the disposal of solid low-level radioactive waste and the storage of transuranic (TRU) radioactive waste and mixed (hazardous and radioactive) waste. The Site also has a central sanitary landfill and buildings in the Central Shops

Table 2-1. Description of functions and principal facilities at SRS areas.

Area	Function	Principal facilities
A	Main DOE administration area, research laboratories	Main administration building, Savannah River Technology Center, Savannah River Ecology Laboratory, powerhouse
B	Wackenhut Services, Inc., administration area (security)	Administration building, WSRC Engineering building, WSRC training buildings
C	One of five SRS reactors	C-Reactor, training facilities, cooling basin
D	Central powerhouse and heavy-water rework	Powerhouse, heavy-water rework facility
E	Waste disposal and storage	Solid Waste Disposal Facility
F	Process plutonium	F-Area Canyon, FB-Line, tank farm
G	Various support functions	Spread throughout the Site: railroad yard, U.S. Forest Service installations
H	Process uranium and tritium	H-Area Canyon, HB-Line, Effluent Treatment Facility, tank farm, Receiving Basin for Offsite Fuels, Consolidated Incineration Facility
K	One of five SRS reactors	K-Reactor, cooling basins, cooling tower
L	One of five SRS reactors	L-Reactor, cooling basins
M	Production of fuel and target assemblies	Slug and target production facilities, effluent treatment facility
N	Receiving	Central Shops
P	One of five SRS reactors	P-Reactor, cooling basins
R	One of five SRS reactors	R-Reactor, cooling basins
S	Process high-level radioactive waste	Defense Waste Processing Facility
TNX	Applied research and development	Analytical laboratory, Defense Waste Processing Technology facilities, various mockups, effluent treatment facilities
Z	Waste treatment and handling	Saltstone facility

(N-Area) for the storage of nonradioactive hazardous wastes and mixed waste. DOE is preparing an EIS on waste management activities at the SRS (DOE 1995a).

The Site contains facilities for processing support and for research and development. These include operational coal-fired powerhouses in A-, D-, and H-Areas that generate electricity and steam.

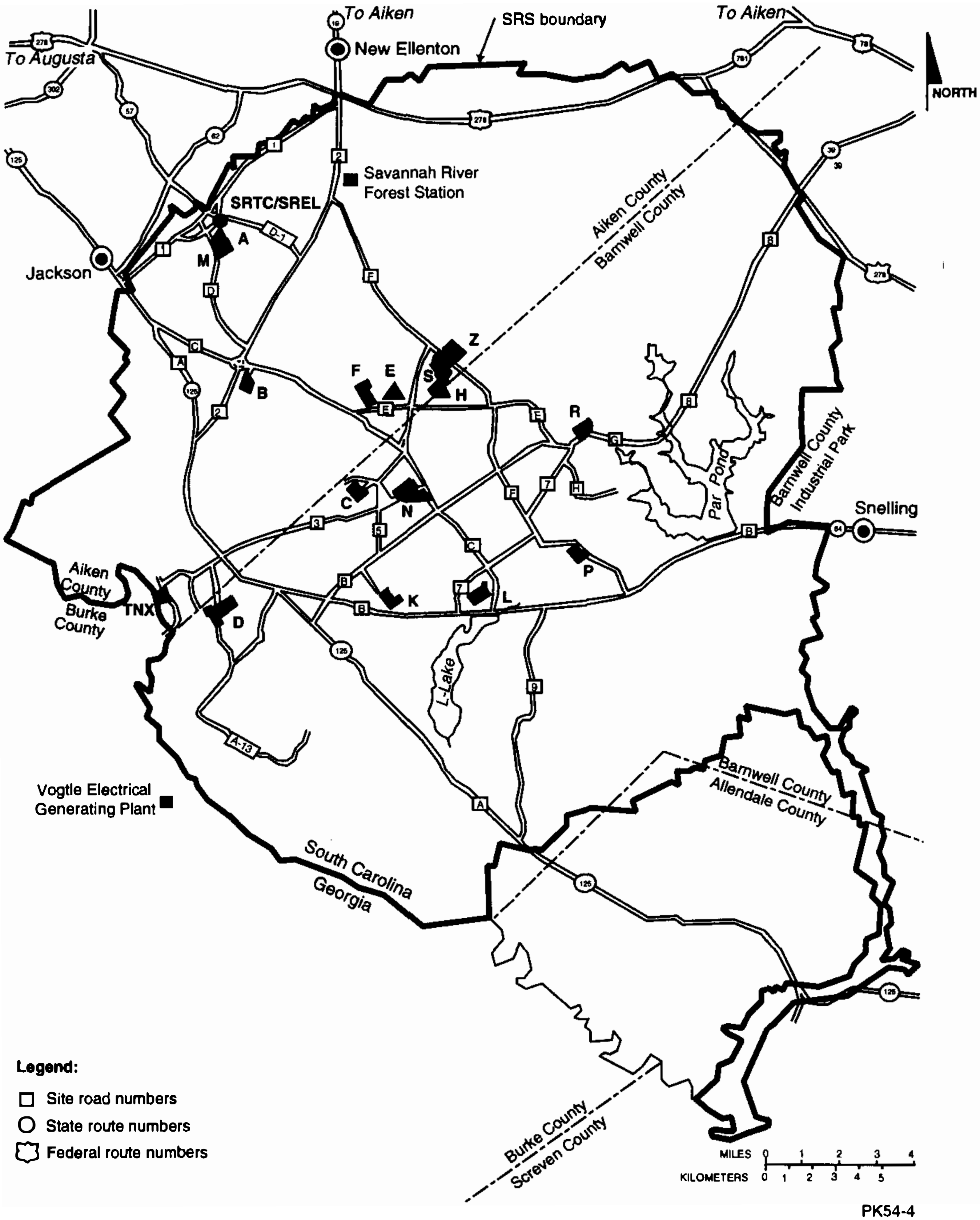

Figure 2-2. Location of principal SRS facilities (see Table 2-1).

The largest powerhouse, which is in D-Area, produces electricity and sends process steam to C-, F-, H-, and S-Areas through a 7-mile (11-kilometer) steam line. D-Area also contains the heavy-water rework facility at which DOE purified the deuterium oxide (heavy water) used as the moderator and coolant in SRS reactors. TNX-Area facilities study chemical and waste processing problems and test production-scale equipment. Finally, A-Area facilities include the Savannah River Technology Center, the Savannah River Ecology Laboratory, and the DOE and Westinghouse Savannah River Company administrative offices.

The SRS employs approximately 20,000 people. Most of these employees work for Westinghouse Savannah River Company and its subcontractors. The remainder work for DOE, the Savannah River Ecology Laboratory, Wackenhut Services, Inc., the U.S. Forest Service, and other contractors.

2.1.2 Site History

The U.S. Atomic Energy Commission (AEC), a DOE predecessor agency, selected the location for the SRS in November 1950 after a study of more than 100 prospective sites. The government selected E. I. du Pont de Nemours and Company, Inc., to build and operate the facility. Construction began in February 1951; the basic plant was completed in 1956 at a cost of $1.1 billion, including the land. On October 3, 1952, operations began with the startup of a unit of the heavy-water extraction plant. Criticality occurred in the first production reactor on December 28, 1953.

In 1972, the AEC designated the SRS as the nation's first National Environmental Research Park. Through the years, scientists have performed a wide range of investigations on the diverse habitats, flora, and fauna of the Site.

2.1.3 Mission

The historic mission of the SRS was to serve the national security interests of the United States by safely processing nuclear materials while protecting the health and safety of employees and the public and protecting the environment. The SRS was responsible for producing tritium and special nuclear materials for national defense. At present, it supports the viability of the weapons stockpile by recycling limited-life components. The SRS also produces isotopes for nonweapons applications in the nation's space program and for medical applications.

The SRS spent nuclear fuel mission is to manage DOE-owned spent fuel in a cost-effective way that protects the safety of SRS workers, the public, and the environment. The goals of near-term activities are the accurate quantification and characterization of DOE-owned spent nuclear fuel, assessment of spent nuclear fuel storage facilities, elimination of current spent nuclear fuel storage vulnerabilities, and identification of technologies and requirements for interim management and ultimate disposition of spent nuclear fuel.

2.1.4 Management

The DOE Savannah River Operations Office manages the SRS; the Westinghouse Savannah River Company operates the Site under contract to DOE. Westinghouse assumed operational responsibility in April 1989 from E. I. du Pont de Nemours and Company, Inc., which had operated the Site since 1951.

2.2 Regulatory Framework

This section summarizes the framework of environmental protection regulations applicable to spent nuclear fuel management at the SRS. The framework is based on Federal and South Carolina laws and one local ordinance, as discussed below. Volume 1 (Section 7.0) of this Environmental Impact Statement (EIS) provides additional information on the major Federal environmental laws and regulations, Executive Orders, and DOE Orders that apply to spent nuclear fuel management alternatives.

2.2.1 Federal

The U.S. Environmental Protection Agency (EPA) has authorized South Carolina to implement most provisions of the Clean Air Act, Resource Conservation and Recovery Act, and Clean Water Act that apply to SRS spent nuclear fuel management. EPA Region IV has the lead responsibility for Clean Air Act standards for radionuclide emissions from DOE facilities, imposing monitoring and approval requirements on SRS spent nuclear fuel management activities that could result in radionuclide emissions.

In addition, EPA Region IV has Resource Conservation and Recovery Act authority over radioactive hazardous (mixed) waste management, affecting wastes from spent nuclear fuel processing.

EPA Region IV and the DOE Savannah River Operations Office have entered into a Federal Facility Compliance Agreement on SRS mixed waste management.

The U.S. Army Corps of Engineers District Engineer for the Charleston District implements the Clean Water Act Section 404 and the Rivers and Harbors Act permitting program for SRS spent nuclear fuel construction activities that would affect U.S. waters.

In accordance with the Endangered Species Act, the SRS would consult with the U.S. Fish and Wildlife Service, Charleston Field Office on impacts that spent nuclear fuel construction activities could have on threatened and endangered species.

2.2.2 State

The South Carolina Department of Health and Environmental Control implements the following State laws that would affect SRS spent nuclear fuel management activities:

- Pollution Control Act (nonradioactive emissions and discharges, and nonhazardous waste management)
- Hazardous Waste Management Act (nonradioactive hazardous waste management)
- Safe Drinking Water Act
- Groundwater Use Act
- Stormwater Management and Sediment Reduction Act

The U.S. Army Corps of Engineers District Engineer for the Charleston District has an agreement with the South Carolina Department of Health and Environmental Control whereby that department issues Clean Water Act Section 401 water quality certifications. The South Carolina Department of Health and Environmental Control also receives SRS reports in accordance with the Emergency Planning and Community Right-To-Know Act.

The South Carolina State Department of Archives and History includes the State Historic Preservation Office. In accordance with the National Historic Preservation Act, the SRS would consult with the State Historic Preservation Officer on impacts that construction activities could have on cultural resources.

2.2.3 Local

The only local requirement applicable to SRS spent nuclear fuel management is the Aiken County Sediment Control Ordinance, which would affect construction activities.

2.3 Spent Nuclear Fuel Management Program at the Savannah River Site

This EIS addresses the management of approximately 2,742 metric tons of heavy metal (MTHM; 3,023 tons) of spent nuclear fuel that would be stored at various locations within the DOE Complex over the next 40 years (1995-2035). At present, DOE has stored approximately 206.3 MTHM (227.4 tons), or about 8 percent of this material, at the SRS. The spent nuclear fuel currently stored at the SRS that DOE has included in the analyses in this document includes:

- 184.4 MTHM (203.3 tons) of Savannah River Defense Production [highly enriched uranium (HEU) aluminum-clad fuels], including plutonium target material, and other aluminum-clad fuels

- 4.6 MTHM (5.1 tons) of commercial spent fuel (primarily zirconium-clad)

- 11.9 MTHM (13.1 tons) of test and experimental reactor Zircaloy-clad fuel

- 5.4 MTHM (6.0 tons) of test and experimental reactor stainless steel-clad fuel

Spent nuclear fuel is currently stored in the Receiving Basin for Offsite Fuels (RBOF), in three reactor disassembly basins, and in basins in F- and H-Canyons. Table 2-2 shows the quantity of spent fuel stored at these facilities.

Table 2-2. SRS Fuel Inventory by Facility.

Facility	Quantity (MTHM)
Receiving Basin for Offsite Fuel	60.73
L-Reactor Disassembly Basin	118.11
K-Reactor Disassembly Basin	3.32
P-Reactor Disassembly Basin	1.41
F-Canyon	22.63
H-Canyon	0.07
Total	206.27

Source: Wichmann (1995).

The F- and H-Area Canyons at the SRS are among the only remaining operable chemical separations facilities of their kind in the DOE Complex. Each canyon has an associated storage basin that serves as an interim staging area where reactor fuel bundles and targets await the Chemical Separations Process. The basins currently contain 13 reactor fuel assemblies (H-Area) and aluminum-clad targets (F-Area).

DOE has stored most of the remaining aluminum-clad spent nuclear fuel from SRS reactor operations under water in concrete reactor storage basins. Three reactor disassembly basins (K-, P-, and L-Reactors) contain reactor fuel and target material. These structures were built in the 1950s and were not intended for the prolonged storage of radioactive materials. Wet (underwater) storage, while potentially viable for stainless steel-clad fuel elements, is not satisfactory for aluminum-clad elements, which are subject to corrosion and pitting.

In March 1992, chemical processing operations were suspended in the canyons to address a potential safety concern. The concern was subsequently addressed but prior to resumption of processing, the Secretary of Energy directed that defense related chemical separations activities (i.e., reprocessing) be phased out at the SRS. Since the decision, DOE has determined that further action related to the disposition of nuclear material, including spent nuclear fuel, is subject to the National Environmental Policy Act (NEPA) process. Non-safety related facility operations have remained shut down with the exception of Pu-238 processing associated with the support of NASA missions.

As a result of these shut-downs, the canyons and the basins used for storage of spent nuclear fuel and irradiated targets have a large inventory of in-process solutions and fuel and targets (respectively).

Some materials stored in the L- and K-Reactor disassembly basins have corroded, releasing fissile materials to the pool water. DOE is preparing an environmental impact statement that will evaluate risks that these and other SRS materials represent to the public and workers and will assess the near-term need for the actions to stabilize these materials to ensure continued safe management (DOE 1995b). These actions would take place over the short-term (about 10 years), until DOE can make programmatic decisions on disposition.

DOE stores other spent fuel in the Receiving Basin for Offsite Fuels (RBOF) on the SRS. This basin, which is in H-Area near the center of the Site, has been operating and receiving fuels of U.S. origin since 1964. This 15,000-square-foot (1,393-square-meter) facility consists of an unloading basin, two storage basins, a repackaging basin, a disassembly basin, and an inspection basin. The basins and their interconnecting transfer canals hold about 500,000 gallons (1,893,000 liters) of water. Spent fuel elements arrive in lead-lined casks weighing from 24 to 70 tons (about 22 to 64 metric tons), which a crane lifts from a railroad car or truck trailer and places in the unloading basin. About 30 percent of the fuels in the Receiving Basin for Offsite Fuels consist of uranium clad in stainless steel or Zircaloy, which SRS facilities cannot process without modifications.

2.4 Vulnerabilities Associated with SRS Spent Nuclear Fuel

In August 1993, the Secretary of Energy commissioned a comprehensive baseline assessment of the environmental, safety, and health vulnerabilities associated with the storage of spent nuclear fuel in the DOE complex. The purpose of this assessment was to determine the inventory and condition of the Department's Reactor Irradiated Nuclear Material, which includes spent nuclear fuel and reactor irradiated target material. The assessment also evaluated the condition of the facilities that store spent fuel and identified the vulnerabilities and problems currently associated with these facilities. Vulnerabilities in nuclear facilities are conditions or weaknesses that could lead to radiation exposure to the public, unnecessary or increased exposure to workers, or release of radioactive materials to the environment. Loss of institutional controls, such as a cessation of facility funding or reductions in facility maintenance and control, could cause some vulnerabilities.

Based on this evaluation process DOE released a report to the Secretary of Energy, entitled *Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and other Reactor Irradiated Nuclear Materials and Their Environmental, Safety and Health*

Vulnerabilities (i.e., "The Working Group Report," Volumes I, II, and III), to the public on December 7, 1993 (DOE 1993). This report identified over 100 vulnerabilities associated with spent fuel storage in the DOE complex, including 19 at the Savannah River Site. The report also determined that five facilities and three burial grounds warranted priority attention from management to avoid unnecessary increases in worker radiation exposure and cost during cleanup. The Savannah River Site L- and K-Reactor Disasssembly Basins were among these facilities. The report grouped vulnerabilities associated with each facility into three categories for management attention based on when corrective action should be initiated: less than 1 year, 1 to 5 years, and more than 5 years.

After issuing the Working Group Report, DOE developed a Plan of Action to address all vulnerabilities, taking into consideration currently available resources for implementation. The Plan of Action is a consolidation of individual action plans designed to address each spent nuclear fuel vulnerability in a manner that reflects the DOE (1) sense of urgency, (2) concern for worker protection, (3) commitment to avoid or otherwise mitigate environmental impacts, and (4) need for compatible long-term solutions.

The interim goal for the Savannah River Site reactor disassembly basins, pending completion of the removal of the stored material, is the stabilization of basin conditions to reduce corrosion and to address known vulnerabilites. The long-term goal of the action plan is a safe start of the removal of reactor-irradiated nuclear material within a 5-year period, consistent with safe and environmentally sound operations, including completion of appropriate NEPA review. These actions will lead to mitigating the identified vulnerabilities while DOE pursues other courses of action.

The 19 vulnerabilities identified for the Savannah River Site now have complete Action Plans (DOE 1994b, 1994c, 1994d). Table 2-3 lists SRS vulnerabilities by facility, tracking number, priority categorization, and Action Plan status.

DOE is currently implementing a number of the 19 Action Plans. These actions have been evaluated under the NEPA review process. The remaining corrective actions, those that will be carried out through FY99, would also undergo NEPA review prior to implementation. Only one of these outstanding actions, the construction of a dry storage facility, would likely require detailed NEPA documentation (e.g., an EIS). The construction of such a facility is addressed programmatically in this EIS as part of the Decentralization, 1992/1993 Planning Basis, Regionalization, and Centralization alternatives. Construction of new facilities would require site-specific NEPA documentation, however.

Table 2-3. SRS vulnerabilities by facility, vulnerability, tracking number, priority categorization, and Action Plan status.

Site/Facility Vulnerability Number Description	Priority Eight major facilities with vulnerabilities	 Less than 1 year	 Greater than 1 year	Action Plan status
SRS/L-Reactor Disassembly Basin SRS-01 Potential unmonitored buildup of radionuclide or fissile materials in sand filters.	✓			Complete
SRS/L-Reactor Disassembly Basin SRS-04 Lack of authorization basis in operating the sand filter cleanup system for L-Area Disassembly Basin.	✓			Complete
SRS/Reactor Disassembly Basins SRS-05 Corrosion of aluminum clad fuel, targets, and components.			✓	Complete
SRS/L-Reactor Disassembly Basins SRS-06 Cesium-137 activity level in L-Basin.	✓			Complete
SRS/L-Reactor Disassembly Basins SRS-07 Determine whether gas bubbles release is a potential hazard above the bucket storage area at L-Reactor.	✓			Complete
SRS/K-, L-, P-Reactors SRS-08 Lack of Reactor Authorization Basis.	✓			Complete
SRS/K-Reactor Disassembly Basins SRS-09 Corrosion of Mark 31 A and B target slugs in K and L disassembly basins.	✓			Complete
SRS/P-Reactor Disassembly Basins SRS-10 Hoist Rod Corrosion		✓		Complete
SRS/K-, L-Reactor Disassembly Basins SRS-11 Reactor Disassembly Basin Safety Analysis Envelope.	✓			Complete
SRS/L-Reactor Disassembly Basin SRS-12 Inadvertent flooding of L-Reactor Disassembly Basin.	✓			Complete
SRS/K-Reactor Disassembly Basin SRS-13 Inadvertent flooding of K-Reactor Disassembly Basin.	✓			Complete
SRS/P-Reactor Disassembly Basin SRS-14 Inadvertent flooding of P-Reactor Disassembly Basin.	✓			Complete

Table 2-3. (continued).

Site/Facility Vulnerability Number Description	Priority: Eight major facilities with vulnerabilities	Priority: Less than 1 year	Priority: Greater than 1 year	Action Plan status
SRS/RBOF; P-, R-, L-, C-, R-Reactors SRS-15 (NOTE: RBOF is a less than 1 year vulnerability) Conduct of operations at reactor facilities and RBOF.	✓			Complete
SRS/Receiving Basin for Offsite Fuel (RBOF) SRS-16 Inadequate tornado protection at RBOF.		✓		Complete
SRS/Receiving Basin for Offsite Fuel (RBOF) SRS-17 Seismic vulnerability of RBOF.		✓		Complete
SRS/H-Area Canyon SRS-18 Seismic vulnerability of H-Area Canyon.			✓	Complete
SRS/F-Area Canyon SRS-19 Seismic vulnerability of F-Area Canyon.			✓	Complete
SRS/K-, L-, P-Reactor Disassembly Basins and RBOF SRS-20 Inadequate leak detection system in the underground water-filled RINM storage basin.		✓		Complete
SRS/L-, K-, P-Reactor Disassembly Basins SRS-21 Inadequate seismic evaluation and potential inadequacies of structures, systems, and components to withstand a design basis event.	✓			Complete

2.5 Representative Host Sites

DOE has identified two SRS areas as representative host sites for potential facilities related to the implementation of programmatic decisions on spent nuclear fuel management (Figure 2-3):

- F- and H-Areas (considered together) for the modification or expansion of existing facilities, new wet storage, and support facilities

- An undeveloped site for the construction of major new facilities, primarily an Expended Core Facility or dry storage vault.

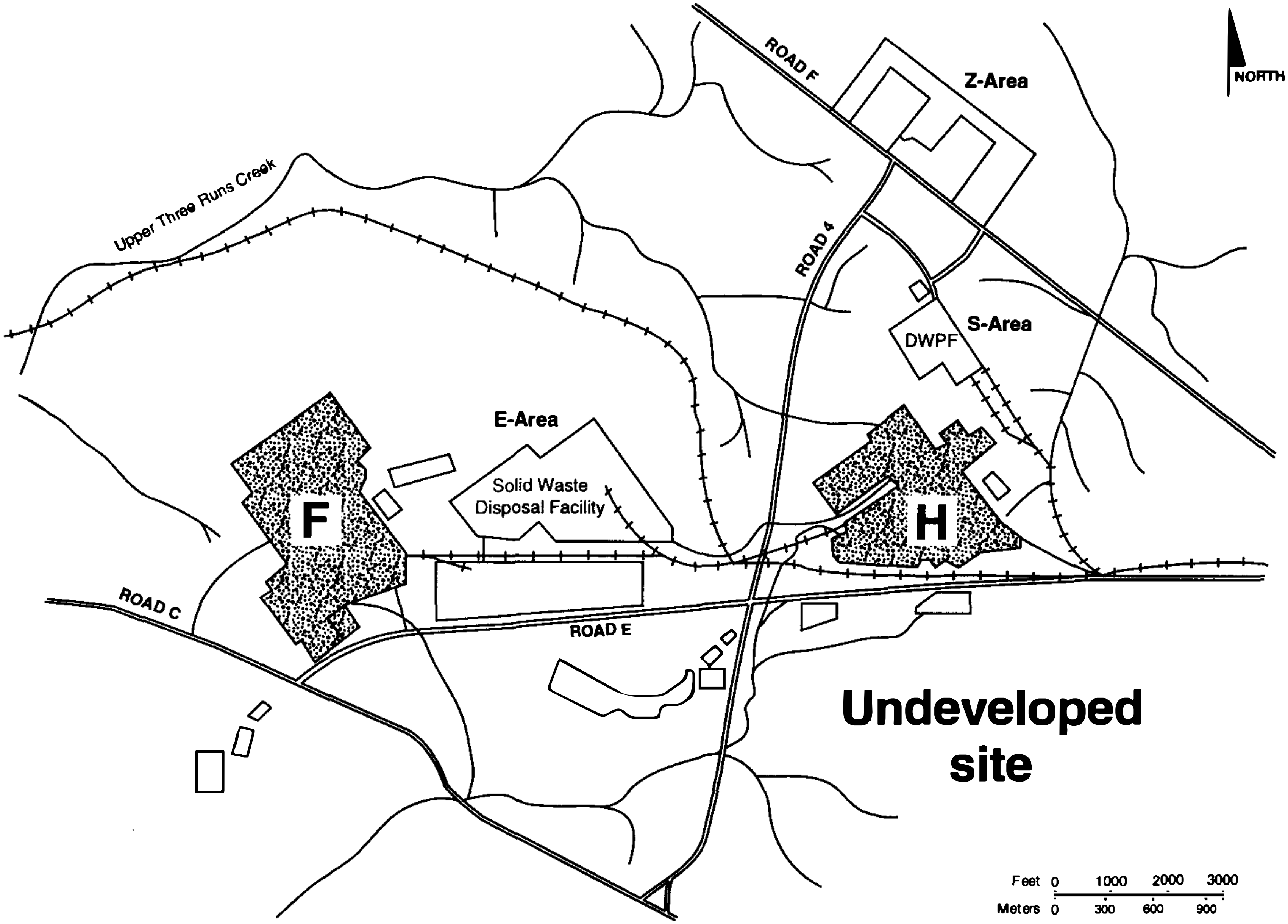

Figure 2-3. Representative host sites on Savannah River Site.

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2.5.1 F- and H-Areas

These two areas contain most of the current spent nuclear fuel facilities and operations at the SRS, including the Receiving Basin for Offsite Fuels. Therefore, DOE would focus future actions under any of the alternatives in these areas as well, for cost-effectiveness and because construction would occur in areas that had been previously disturbed.

F- and H-Areas are about 2 miles (3.2 kilometers) apart near the center of the SRS. The nearest Site boundary is approximately 7.5 miles (12 kilometers) to the west. DOE uses the land within a 5-mile (8-kilometer) radius of the two areas either for industrial purposes associated with SRS operations or as managed forest land. The closest facility to F- and H-Areas is the E-Area Solid Waste Disposal Facility, which lies between the two areas (Figure 2-3). DOE uses this facility to dispose of SRS solid low-level radioactive waste and to store TRU radioactive waste and mixed waste.

The F-Area separations facilities occupy about 420 acres (1.7 square kilometers). These facilities were designed primarily for the recovery of plutonium-239 from irradiated and unirradiated feed materials. DOE used the F-Area Canyon to dissolve target materials and produce solutions that contained the various products extracted from fission products. Further processing converted the products from solution to solid form for shipment off the Site. Large tanks in F-Area store high-level liquid radioactive waste for future stabilization and disposal through the Defense Waste Processing Facility.

H-Area facilities occupy about 395 acres (1.6 square kilometers). The H-Area Canyon processed irradiated fuel elements or target assemblies from reactors. Primary operations included the dissolution of irradiated targets and fuel tubes, chemical and physical separation, and purification of materials. DOE stores high-level liquid waste in large tanks in H-Area, as in F-Area, for future processing and disposal through the Defense Waste Processing Facility.

2.5.2 Undeveloped Representative Host Site

DOE has selected an undeveloped representative host site for the construction of new facilities that F- or H-Area could not accommodate. This site is to the south and east of H-Area, adjacent to SRS Road E and close to an existing railroad line, as shown in Figure 2-3. The SRS could make connections to existing electricity, water, and steam networks with minimal additional construction.

The use of this site would have the advantage of consolidating spent nuclear fuel-related activities near F- and H-Areas and close to the center of the SRS.

This site is representative of many available areas on the SRS that could support spent nuclear fuel management activities. For example, DOE has identified a different representative site for the possible construction of the Expended Core Facility for the management of naval spent nuclear fuel (see Appendix D of Volume 1 of this Environmental Impact Statement). DOE would conduct a detailed siting analysis before implementing any programmatic decision at the SRS. DOE would assess, as necessary, the environmental consequences of the siting of any facilities as part of the site-specific NEPA documentation.

3. SPENT NUCLEAR FUEL ALTERNATIVES

This chapter describes the five management alternatives for spent nuclear fuel that the U.S. Department of Energy (DOE) has evaluated for the Savannah River Site (SRS) as part of Volume 1 of this Environmental Impact Statement. These alternatives are:

1. No Action
2. Decentralization
3. 1992/1993 Planning Basis
4. Regionalization (with 2 subalternatives for the SRS)
5. Centralization (with 2 subalternatives for the SRS)

The activities covered by the alternatives range from maintaining the current inventory of spent fuel at the SRS without receiving any more shipments (Alternative 1), through keeping the existing inventory and accepting or sending off some limited shipments (Alternatives 2 through 4), to receiving at the Site all DOE spent nuclear fuel and some from other sources (Alternative 5). DOE also examined an option for shipping all spent nuclear fuel at the SRS to another location (a variation of Alternatives 4 and 5). Table 3-1 summarizes the quantities of material that would be received, shipped out, and ultimately managed at the SRS under the various alternatives. DOE has assessed the aluminum-clad spent nuclear fuel separately from nonaluminum-clad fuel (i.e., stainless steel and Zircaloy) because the options for managing them at the Site could be different as explained in Section 3.1.

The analytical approach used in this document produces estimates of consequences that would be as large as or larger than any that could occur or be expected under the alternatives and provides a comparison of the impacts of the principal technologies for managing spent nuclear fuel at the SRS.

This chapter also provides an overview of the SRS management approach and describes the five alternatives as they relate to the SRS (Sections 3.1 and 3.2). In addition, the chapter summarizes and compares the potential environmental consequences of each alternative (Section 3.3).

Table 3-1. Quantities (MTHM)[a] of spent nuclear fuel that would be received, shipped, and managed at the SRS under the five alternatives.[b,c]

Alternative	Fuel Type	Currently at SRS	Receive	Ship Out	Totals managed at SRS under this alternative
1. No Action	Aluminum	184.40	0.00	0.00	184.40
	Nonaluminum	21.87	0.00	0.00	21.87
	Totals	206.27	0.00	0.00	206.27
2. Decentralization	Aluminum	184.40	11.02	0.00	195.42
	Nonaluminum	21.87	2.60	0.00	24.47
	Totals	206.27	13.62	0.00	219.89
3. 1992/1993 Planning Basis	Aluminum	184.40	13.69	0.00	198.09
	Nonaluminum	21.87	2.80	0.00	24.67
	Totals	206.27	16.49	0.00	222.76
4. Regionalization - A (by fuel type)	Aluminum	184.40	28.69	0.00	213.09
	Nonaluminum	21.87	0.00	(21.87)	0.00
	Totals	206.27	28.69	(21.87)	213.09
4. Regionalization - B (by location at SRS)	Aluminum	184.40	19.93	0.00	204.33
	Nonaluminum	21.87	30.42	0.00	52.29
	Totals	206.27	50.35	0.00	256.62
4. Regionalization - B (by location, elsewhere)	Aluminum	184.40	0.00	(184.40)	0.00
	Nonaluminum	21.87	0.00	(21.87)	0.00
	Totals	206.27	0.00	(206.27)	0.00
5. Centralization (at SRS)	Aluminum	184.40	28.69	0.00	213.09
	Nonaluminum	21.87	2,506.84	0.00	2,528.71
	Totals	206.27	2,535.53	0.00	2,741.80
5. Centralization (elsewhere)	Aluminum	184.40	0.00	(184.40)	0.00
	Nonaluminum	21.87	0.00	(21.87)	0.00
	Totals	206.27	0.00	(206.27)	0.00

a. To convert metric tons of heavy metal to tons, multiply by 1.1023.
b. Numbers may not sum due to rounding.
c. Source: Wichmann (1995).

3.1 SRS Management Approach

3.1.1 Management Options

DOE has evaluated three options for the management of spent nuclear fuel at the SRS under the five alternatives considered for this EIS. These technical management options are wet storage or dry storage of all fuels and the processing of aluminum-clad fuels. DOE could implement these options individually or in combination under any of the five alternatives. DOE would base its selection of one or more of these technical management options on additional analysis, including a separate SRS-specific National Environmental Policy Act (NEPA) review based on this programmatic EIS.

3.1.1.1 Wet Storage. As described above in Section 2.3, the SRS currently maintains its spent nuclear fuel in wet storage in the Receiving Basin for Offsite Fuels and several reactor basins. Wet storage under the 40-year interim management plan (except under the No Action alternative) would require that DOE construct a new wet storage pool at the SRS and move all fuel to this facility. Prior to this transfer, DOE could place all the aluminum-clad fuel in stainless steel canisters to prevent further corrosion and breakdown of the fuel cladding. The stainless steel- and Zircaloy-clad fuels could also require canning. The SRS would monitor and maintain the water quality and the condition of the fuel in the storage pool throughout the interim management period.

Under this wet storage option, the spent nuclear fuel would be in an interim storage form, which could require further treatment depending on the DOE decision on its ultimate disposition.

3.1.1.2 Dry Storage. DOE currently has no dry storage facilities for spent nuclear fuel at the Site. Dry storage of SRS aluminum-clad fuels under this management plan would require technology development prior to the construction of a dry storage facility. Although such facilities exist at other DOE sites and at commercial locations, DOE believes that the characteristics of SRS spent fuel are sufficiently different to require some research and development before the design and construction of a facility for this fuel. DOE would can all fuel before placing it into the dry storage vaults. It would also have to maintain and monitor the facility for the remainder of the 40-year management period.

As with wet storage, the dry storage option would place the spent fuel into an interim storage form that could require further treatment later depending upon DOE's decision on ultimate disposition.

3.1.1.3 Processing and Dry Storage. One method under this option would be for the SRS to process existing aluminum-clad spent nuclear fuel through the existing separations facilities in the F- and H-Area Canyons, and place the nonaluminum-clad fuels and any future receipts in dry storage. The process using existing capability would result in the generation of both separated actinides (e.g., uranium oxide), which would be stored on the site in existing facilities, and solutions of fission products that would be placed in existing waste storage facilities for later conversion to a glassified form through the Defense Waste Processing Facility (DWPF). DOE would maintain and monitor the dry storage facility containing the nonaluminum-clad spent fuel. Variations of this processing option are also possible, such as processing all the aluminum-clad fuel currently on the Site plus all that is received from elsewhere, or developing the capability at the SRS for processing for vitrification without chemical separations.

The process option selected for evaluation in this document is representative of possible processing options that might be employed, but is not necessarily the one that DOE would select. Detailed NEPA evaluations would be required to implement any spent nuclear fuel management plan at the SRS.

3.1.2 Management Plan

Figure 3-1 summarizes DOE's overall plan for the interim management of aluminum-clad and nonaluminum-clad fuels at the SRS. This flowchart shows actions for all alternatives except No Action, as explained in Section 3.2.1.

3.1.2.1 Aluminum-clad Fuels. Depending on the alternative and option selected, DOE could (within constraints of mission commitments) consolidate some aluminum-clad fuel in the Receiving Basin for Offsite Fuels to take advantage of this facility's superior water quality and then move all aluminum-clad fuel into dry storage, wet storage, or initiate processing (Figure 3-1). DOE could also process aluminum-clad fuel without any consolidation work. Before moving the fuel into dry or wet storage, DOE would place it in cans. DOE would hold the canned fuel or the stabilized products from processing in storage for the 40-year interim management period until it decided their final disposition.

DOE would place aluminum-clad fuels received by the SRS from other locations in wet or dry storage. DOE could not implement any of the options for aluminum-clad fuels, with the exception of processing using existing SRS capabilities, without a technology development effort.

3.1.2.2 Nonaluminum-clad Fuels. DOE options for the management of nonaluminum-clad fuels at the SRS are somewhat different, in that only dry or wet storage is considered (Figure 3-1). The processing of these fuels at the Site is not an option because the SRS does not currently have operational facilities capable of separating these materials. To improve aluminum-clad fuel storage, DOE could consolidate the nonaluminum-clad fuel inventory in a reactor basin where the more resistant stainless steel or Zircaloy cladding would be less susceptible to corrosion. The fuel would remain there until DOE built new dry or wet storage facilities. DOE would then can the fuel and move it into the new storage. DOE would place any nonaluminum-clad fuel received at the SRS after completion of the new facilities directly into storage. The fuel would remain in this interim storage until DOE decided its ultimate disposition.

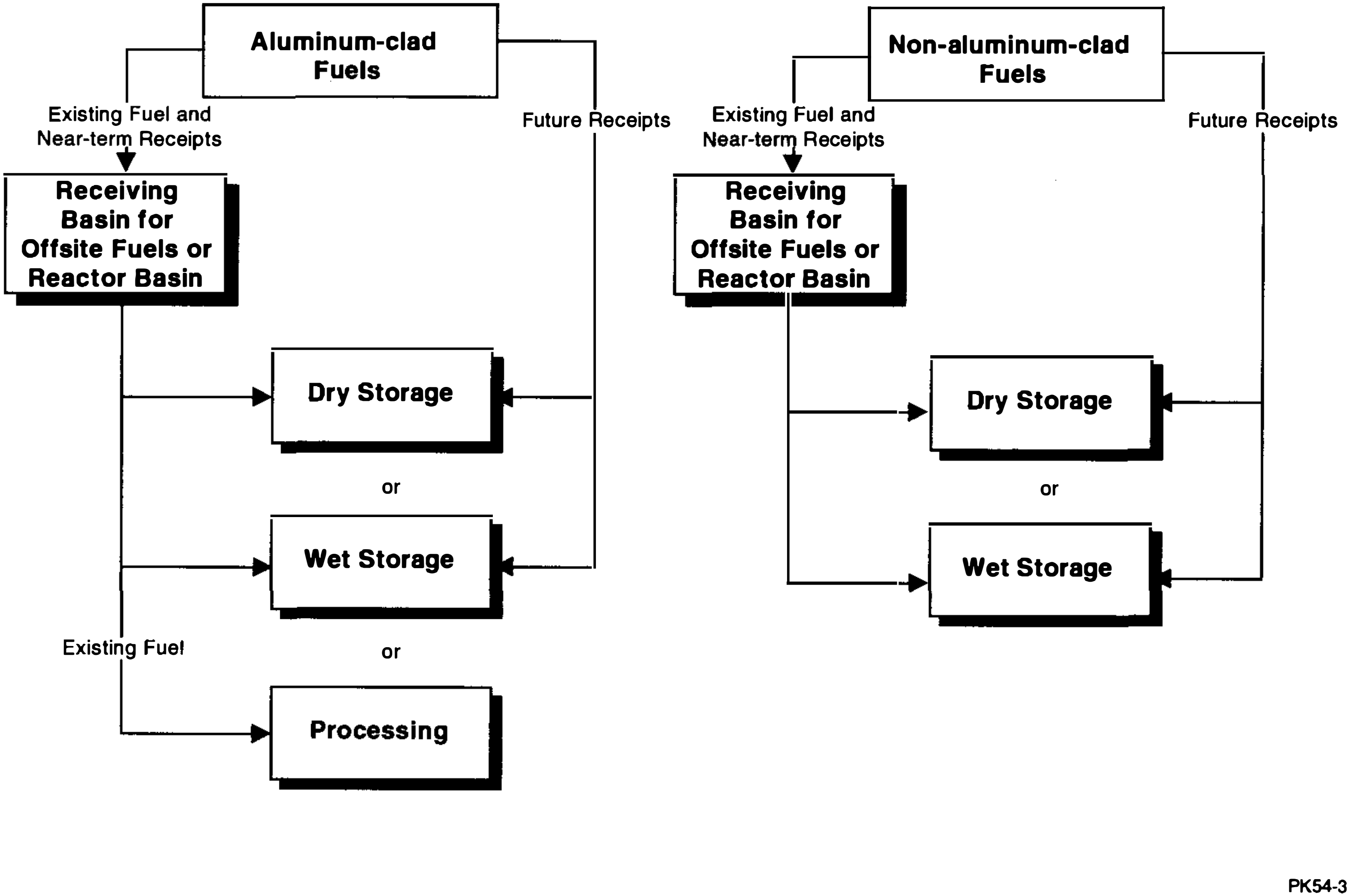

Figure 3-1. Diagram of how SRS would manage aluminum-clad and nonaluminum-clad fuels. "Near-term Receipts" refers to the fuel that would be received before new wet or dry storage facilities are available.

3.2 Description of Alternatives

3.2.1 Overview

Table 3-2 compares actions under each of the five alternatives. These actions relate to the requirements for transportation, stabilization, facilities, and research and development that DOE would address for each alternative. Transportation would include onsite movements as well as the receipt or shipment of spent fuel. The consideration of facilities addresses not only new ones that could be required, but also the use of existing structures and capabilities such as the F- and H-Area Canyons at SRS. Finally, each alternative would involve some level of research and development on matters related to spent nuclear fuel interim management (e.g., stabilization, transportation casks) and its ultimate disposition.

Alternative 1 (No Action) addresses only the interim wet storage option, while the analysis of Alternatives 2 through 5 considers three options: dry storage, wet storage, and processing of existing aluminum-clad fuels and placing the other fuels into storage. In addition, Alternatives 4 and 5 include an option for the shipment of spent nuclear fuel off the SRS. This analytical approach shows the relative impact of viable interim storage technologies for the range of alternatives this EIS is considering for the SRS. However, this information is not sufficient to support the selection of a specific interim storage technology at the SRS because DOE has not completed site-specific research and development for dry storage and wet storage methods or an evaluation of other processing options. In addition, the specific quantities of offsite fuel that DOE would manage are subject to change. The selection of an interim storage technology will be the subject of separate NEPA documentation specific to the SRS.

Figure 3-2 is a matrix showing the types of facilities that would be required for each alternative and option. The list includes those facilities already operating at the SRS (e.g., Receiving Basin for Offsite Fuels) as well as potential facilities (e.g., fuel characterization facility). DOE considered these facilities in its evaluation of the consequences of each alternative, as described in Chapter 5.

The alternatives described below address interim storage to 2035; further treatment of the spent nuclear fuel would be necessary before DOE obtained a final disposable waste form. This EIS does not address this additional treatment. However, DOE would carry out a full NEPA documentation for any decision on final disposition of spent nuclear fuel.

Table 3-2. Actions required under each of the five alternatives at the SRS.

Alternative	Transportation	Stabilization	Facilities	Research and Development
1. No Action	No shipments to or from the Site. Limit onsite transfers to those required for safe storage.	Place aluminum-clad fuels that are badly corroded and in danger of cladding failure in containers and return them to wet storage.	Store fuels in Receiving Basin for Offsite Fuels and in an upgraded reactor basin. Requires no new facilities.	Continue existing spent nuclear fuel-related research and development.
2. Decentralization	Receive about 13.6 MTHM (15.0 tons) of aluminum-clad and nonaluminum-clad fuels. Limit onsite transfers to those required for safe storage, consolidation, and research and development. Later relocate fuels to new wet or dry storage facility or move aluminum-clad fuels to F- and H-Canyons for processing.	Can aluminum-clad fuels and place them in wet or dry storage or process existing fuel through F- and H-Canyons. Can stainless-steel and Zircaloy fuels and place in wet or dry storage.	Store fuels in Receiving Basin for Offsite Fuels or upgraded reactor basin until new wet or dry storage facility is built. Requires new characterization facility, new wet or dry canning facility, and new wet or dry storage facility.	Develop technology (canning and storage design) to store SRS aluminum-clad fuels in dry storage vault. Conduct research and pilot-scale operations to determine best technology for ultimate disposition of aluminum-clad fuels.
3. 1992/1993 Planning Basis	Receive about 16.5 MTHM (18.2 tons) of aluminum-clad and nonaluminum-clad fuels. Limit onsite transfers to those required for safe storage, consolidation, and research and development. Later relocate fuels to new wet or dry storage facility, or move aluminum-clad fuels to F- and H-Canyon for processing.	Can aluminum-clad fuels and place them in wet or dry storage or process existing fuel through F- and H-Canyons. Can stainless steel and Zircaloy fuels and place in wet or dry storage.	Store fuels in Receiving Basin for Offsite Fuels or upgraded reactor basin until new wet or dry storage facility is built. Requires new characterization facility, new wet or dry canning facility and new wet or dry storage facility.	Develop technology (canning and storage design) to store SRS aluminum-clad fuels in dry storage vault. Conduct research and pilot-scale operations to determine best technology for ultimate disposition of aluminum-clad fuels.
4. Regionalization - A (by fuel type at the SRS)	Receive about 28.7 MTHM (31.6 tons) of aluminum-clad fuel. Ship to Idaho National Engineering Laboratory about 21.9 MTHM (24.1 tons) of stainless steel and Zircaloy fuel. Relocate aluminum-clad fuels to Receiving Basin for Offsite Fuels, as necessary; then to new wet or dry storage facilities, or move aluminum-clad fuels to F- and H-Canyon for processing.	Can aluminum-clad fuels and place them in wet or dry storage; or process existing fuel through F- and H-Canyons.	Store fuel in existing Receiving Basin for Offsite Fuels or upgraded reactor basin until new wet or dry storage facility is available, or until fuel is processed. Requires new receiving and characterization facilities, new wet or dry canning facilities, and new wet or dry storage facilities.	Develop technology (canning and storage design) to store aluminum-clad fuels in dry storage vault. Conduct research and pilot-scale operations to determine best technology for ultimate disposition of aluminum-clad fuels.

Table 3-2. (continued).

Alternative	Transportation	Stabilization	Facilities	Research and Development
4. Regionalization - B (by location at the SRS)	Receive approximately 50.4 MTHM (55.6 tons) of spent fuel from other locations. Limit onsite transfers to those required for safe storage, consolidation, and research and development. Relocate fuels to new dry or wet storage facility or move aluminum-clad fuel to F- and H-Canyons for processing.	Can aluminum-clad fuels and place them in wet or dry storage; or process existing aluminum-clad fuels through F- and H-Canyons and store remaining fuel. Characterize and can fuel received from offsite that is not in a form suitable for direct placement into storage.	Store fuels in Receiving Basin for Offsite Fuels or upgraded reactor basin until new storage facility is available. Store new fuel shipments in new wet or dry storage facility. Requires new receiving, characterization and canning facilities, new wet or dry storage facility, and possibly a new Expended Core Facility.	Develop technology (canning and storage design) to store SRS aluminum-clad fuels in dry storage vault. Conduct research and pilot-scale operations to determine best technology for ultimate disposition of aluminum-clad fuels.
4. Regionalization - B (by location at another site)	Move all fuels to new characterization facility prior to shipment offsite. Ship out about 206.3 MTHM (227.4 tons) of spent fuel.	Characterize and can all spent fuel prior to shipment.	Store existing fuels in Receiving Basin for Offsite fuel and in a reactor basin until characterization and shipment offsite. Requires new characterization facility.	Develop technology for stabilization, canning, and shipment of degraded aluminum-clad fuel.
5. Centralization (at the SRS)	Receive about 2,535.5 MTHM (2,794.9 tons) of spent fuel from offsite. Limit onsite transfers to those required for safe storage, consolidation, and research and development. Relocate fuels to new dry or wet storage facility or move aluminum-clad fuel to F- and H-Canyons for processing.	Can aluminum-clad fuels and place them in wet or dry storage; or process existing aluminum-clad fuels through F- and H-Canyons and store remaining fuels. Characterize and can fuel received from offsite that is not in a form suitable for direct placement in storage.	Store fuel in Receiving Basin for Offsite Fuels or in an upgraded reactor basin until new storage facilities are available. Store new fuel shipments in new wet or dry storage facility. Requires new receiving, characterization and canning facilities, new wet or dry storage facility, and new Expended Core Facility.	Develop technology (canning and storage design) to store SRS aluminum-clad fuels in dry storage vault. Conduct research and pilot-scale operations to determine best technology for ultimate disposition of spent nuclear fuels.
5. Centralization (at another site)	Move all fuels to new characterization facility prior to shipment offsite. Ship out about 206.3 MTHM (227.4 tons) of spent fuel.	Characterize and can all spent fuel prior to shipment.	Store existing fuel in Receiving Basin for Offsite Fuel or in an upgraded reactor basin until characterization and shipment offsite. Requires new characterization facility.	Develop technology for stabilization, canning, and shipment of degraded aluminum-clad fuel.

Facility	No Action	Decentralization			1992/93 Planning Basis			Regionalization - A (by Fuel Type)		
	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c
	Wet	Dry	Wet	Process[c]	Dry	Wet	Process[c]	Dry	Wet	Process[c]
Reactor Basins	●	●	●	●	●	●	●	●	●	●
Receiving Basin Offsite Fuels	●	●	●	●	●	●	●	●	●	●
New Fuel Characterization		○	○	○	○	○	○	○	○	
New Dry Canning		○		○	○		○	○		
New Interim Dry Storage		○		○	○		○	○		
New Expended Core (Navy)										
New Fuel Receiving		○	○	○	○	○	○	○	○	○
New Wet Canning[b]			○			○			○	○
New Fuel Storage Pool			○			○			○	○
H-Canyon/H-Area Separations	X			●			●			●
F-Canyon/F-Area Separations	X			●			●			●

Facility	Regionalization - B (by Location)				Centralization			
	Option 4d	Option 4e	Option 4f	Option 4g	Option 5a	Option 5b	Option 5c	Option 5d
	Dry	Wet	Process[c]	Ship	Dry	Wet	Process[c]	Ship
Reactor Basins	●	●	●	●	●	●	●	●
Receiving Basin Offsite Fuels	●	●	●	●	●	●	●	●
New Fuel Characterization	○	○	○	○	○	○	○	○
New Dry Canning	○		○		○		○	
New Interim Dry Storage	○		○		○		○	
New Expended Core Facility (Navy)	✱	✱	✱		○	○	○	
New Fuel Receiving	○	○	○		○	○	○	
New Wet Canning[b]		○				○		
New Fuel Storage Pool		○				○		
H-Canyon/H-Area Separations			●				●	
F-Canyon/F-Area Separations			●				●	

Legend:

○ New facilities required under each case

● Existing facilities required under each case

X Existing facilities that would be involved to maintain safe storage

✱ May be needed

a. Information derived from WSRC (1994).

b. Includes fuel repackaging facility.

c. Option includes processing of existing aluminum-clad fuels and storage of others.

Figure 3-2. Types of facilities required for each alternative.[a]

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3.2.2 Alternative 1 - No Action

3.2.2.1 Overview. This alternative deals only with the minimum actions that DOE would deem necessary for the continued safe and secure management of spent nuclear fuel. It is not a *status quo* condition. Rather, across its complex of facilities, DOE would maintain spent nuclear fuel close to generation or current storage locations with no shipment between sites. Facility upgrades or replacements and onsite fuel transfers would occur only to support safe and secure interim storage. DOE would continue existing and new research and development activities for spent fuel interim management. Stabilization activities would be limited only to those minimum actions required to store spent nuclear fuel safely.

3.2.2.2 SRS Alternative 1 - Wet Storage. DOE would initiate the various SRS programs and activities necessary to obtain optimum use of existing spent nuclear fuel facilities for the extended storage of existing Site inventories totalling 206.3 metric tons (227.4 tons) of heavy metal (MTHM) in the following quantities:

- 184.4 MTHM (203.3 tons) of Savannah River Defense Production [highly enriched uranium (HEU) aluminum-clad fuels], including plutonium target material, and other aluminum-clad fuels

- 4.6 MTHM (5.1 tons) of commercial spent nuclear fuel (primarily zirconium-clad)

- 5.4 MTHM (6.0 tons) of test and experimental reactor stainless steel-clad fuel

- 11.9 MTHM (13.1 tons) of test and experimental reactor Zircaloy-clad fuel

The goal of this program would be to relocate some aluminum-clad fuels to the Receiving Basin for Offsite Fuels where precisely maintained water quality would prolong the storage life of these fuel types. In addition, DOE would relocate a portion of the stainless steel- and Zircaloy-clad fuels to a reactor basin, where their more resistant cladding would maintain fuel containment for an extended period. These actions would be accomplished within the constraints of mission requirements.

The following describes one method that could be employed to improve the storage of aluminum-clad fuel. Variations of this plan that would involve only the use of existing storage basins are also possible.

- Select a reactor basin for upgrading and for the interim storage of SNF.

- Relocate aluminum-clad fuels from the selected reactor basin to other onsite basins to enable cleaning and repair of the basin chosen for upgrade to improve water quality.

- Consolidate fuels in the Receiving Basin for Offsite Fuels to the extent possible.

- After cleaning and renovating the selected reactor basin, move a portion of the stainless steel and Zircaloy-clad fuel assemblies now at the Receiving Basin for Offsite Fuels to the renovated reactor basin.

- Move the aluminum-clad fuels temporarily stored at other locations to the Receiving Basin for Offsite Fuels or the renovated reactor basin.

DOE will continue to place heavily corroded aluminum-clad fuel elements that could be in danger of cladding failure into containers in the wet pool as required to minimize any spread of materials throughout the pool. This action would be much simpler than canning the elements, which would occur under the other alternatives.

This alternative would require no new facilities. DOE would continue existing spent nuclear fuel-related research and development.

3.2.3 Alternative 2 - Decentralization

3.2.3.1 Overview. Under this alternative, DOE would maintain existing spent nuclear fuel in storage at the current locations, and the SRS would receive some shipments of university fuel and foreign fuel. This alternative differs from the No Action alternative by allowing significant facility development and upgrades. DOE could transport fuel on the Site for safety, fuel consideration, or research and development activities. In addition, DOE could undertake actions it deemed desirable, though not essential, for safety and could perform spent nuclear fuel processing, treatment, research, and development.

3.2.3.2 SRS Options 2a, 2b, and 2c. DOE analyzed three options specific to the SRS for this alternative: Option 2a deals with dry storage, Option 2b deals with wet storage, and Option 2c involves processing existing SRS aluminum-clad spent nuclear fuel and storing the remaining fuel. The amount of spent fuel that the SRS would manage includes its current inventory, as described above for Alternative 1, plus:

- 11.0 MTHM (12.0 tons) of aluminum-clad fuel
- 1.1 MTHM (1.2 tons) of stainless steel-clad fuel
- 0.7 MTHM (0.8 ton) of Zircaloy-clad fuel
- 0.8 MTHM (0.9 ton) of other experimental fuel

Under this alternative, SRS would manage a total of about 219.9 MTHM (242.4 tons) of spent nuclear fuel. The SRS would receive spent fuel from research reactors as existing storage allowed and as new storage was constructed.

3.2.3.2.1 Option 2a - Dry Storage — Under this option, DOE would store existing SRS inventories in wet pools while developing the technology and constructing the necessary facilities to examine, characterize, and can the fuels and transfer them to a new dry storage vault to await treatment for final disposition. The SRS would proceed with the fuel rearrangement plan described above for Alternative 1 to provide acceptable storage conditions to minimize failures of the aluminum-clad material before its placement in a dry-storage container.

Placement in a dry-storage facility would require a technology development program into DOE capabilities to examine, characterize, and can aluminum-clad fuel elements before placing them in a vault. In addition, the SRS would investigate technologies for the ultimate disposition of spent nuclear fuel. In addition to a dry storage facility, the SRS would build new fuel receiving, characterization, and dry canning facilities.

3.2.3.2.2 Option 2b - Wet Storage — Under this option, DOE could rearrange existing spent nuclear fuel as described above for Alternative 1 to provide interim wet storage capacity while constructing new facilities. SRS could also modify this rearrangement plan to accept shipments of spent fuel from offsite and place them directly into the Receiving Basin for Offsite Fuels, as circumstances warrant. The new wet storage facilities required under this option would include the capability to examine and characterize fuels and to can deteriorating fuels in a stainless steel package for placement in the new pool. DOE would move all fuel to the new storage pool once it was complete. SRS would build new fuel receiving, characterization, and wet-canning facilities as well as a new wet storage pool. SRS would investigate technologies for the ultimate disposition of spent nuclear fuel.

3.2.3.2.3 Option 2c - Processing and Storage — Under this option, SRS would process existing aluminum-clad spent nuclear fuel to consolidate and stabilize the nuclear material for storage in vaults, and would place the stainless steel- and Zircaloy-clad fuel and new receipts of aluminum-clad fuel in dry storage. The fuel would remain in the current wet pools while awaiting processing or the construction of new dry storage facilities. DOE would use existing F- and H-Area facilities to process the aluminum-clad fuel to safe, stable, consolidated forms.

The new facilities that the SRS would require under this option would be similar to those described for dry storage (Option 2a), except they would be much smaller because the amount of fuel to be stored would be small: only about 11.0 MTHM (12.0 tons) of aluminum-clad and about 24.5 MTHM (27.0 tons) of nonaluminum-clad fuel.

The SRS would investigate technologies required for the ultimate disposition of spent fuel.

3.2.4 Alternative 3 - 1992/1993 Planning Basis

3.2.4.1 Overview. This alternative assumes the continued transportation, receipt, processing, and storage of spent nuclear fuel. Foreign and university research reactor spent nuclear fuel would be sent to the INEL and the SRS. DOE would assess the construction of new facilities required to accommodate current and projected spent nuclear fuel storage requirements. This alternative would include activities related to the treatment of spent nuclear fuel, including research and development and pilot programs to support future decisions on its ultimate disposition.

3.2.4.2 SRS Options 3a, 3b, and 3c. DOE analyzed the same three options for this alternative as for Alternative 2: dry storage (Option 3a), wet storage (Option 3b), and the processing of existing SRS aluminum-clad fuel and storing the remaining fuel (Option 3c). The quantities of fuel would be somewhat greater than those for Alternative 2 because the options assume that the SRS would manage its present inventory (see Alternative 1) plus approximately:

- 13.7 MTHM (15.1 tons) of aluminum-clad fuel
- 1.3 MTHM (1.4 tons) of stainless steel-clad fuel

- 0.7 MTHM (0.8 ton) of Zircaloy-clad fuel
- 0.8 MTHM (0.9 ton) of other experimental fuel
- a small amount (<0.1 ton) of commercial nonaluminum-clad fuel

The total spent nuclear fuel managed would equal about 222.8 MTHM (245.6 tons). The Site would receive shipments of fuel from other locations as existing space allowed and as new facilities were completed.

3.2.4.2.1 Option 3a - Dry Storage — The Site would store current inventories in existing wet pools while developing technology and constructing facilities necessary to examine, characterize, and can the fuels and transfer them to a new dry storage vault to await treatment for final disposition.

The actions that SRS would undertake under this option and the new facilities to be constructed would be the same as those described for Option 2a - Dry Storage under Alternative 2 (Decentralization) in Section 3.2.3.2.1.

3.2.4.2.2 Option 3b - Wet Storage — DOE could rearrange existing spent nuclear fuel as described in Alternative 1 above to provide interim wet storage capacity while building new facilities. The Site could also accept new shipments directly into the Receiving Basin for Offsite Fuels, as required. The actions that SRS would undertake under this option, and the new facilities to be constructed, would be the same as those described for Option 2b - Wet Storage under Alternative 2 (Decentralization) in Section 3.2.3.2.2.

3.2.4.2.3 Option 3c - Processing and Storage — Under this option, the SRS would process existing aluminum-clad spent nuclear fuel and would place the stainless steel- and Zircaloy-clad fuel and new receipts of aluminum-clad fuel in storage as described for Option 2c - Processing under Alternative 2 (Decentralization) in Section 3.2.3.2.3. The requirements for new facilities and for technology development would also be the same.

3.2.5 Alternative 4 - Regionalization

3.2.5.1 Overview. This alternative has two subalternatives. The first (Regionalization A) would involve the distribution of existing and new spent nuclear fuel among DOE sites based primarily on the similarity of fuel type, although DOE would also consider transport distances, available processing capabilities, available storage capabilities, or a combination of these factors. Under this subalternative, SRS would receive all aluminum-clad fuel and would transfer its existing inventory of stainless steel- and Zircaloy-clad fuel to another DOE site. The SRS would manage a total of about 213.1 MTHM (234.9 tons) of spent fuel under the Regionalization A subalternative.

The second subalternative (Regionalization B) would require DOE to consolidate all existing and new spent fuel at two sites — one to the east of the Mississippi River and one to the west — depending on the location or generation site of the fuel. Under this alternative, the SRS would either receive all spent nuclear fuel in the east [approximately 256.6 MTHM (282.9 tons)] or ship its current inventory offsite to the Oak Ridge Reservation in Tennessee. An additional option if SRS becomes the Eastern Regional Site is for DOE to construct an Expended Core Facility at the SRS to manage some Naval fuel. This option is described in Appendix D of Volume 1 of this EIS.

Under either subalternative, DOE would undertake facility upgrades, replacements, and additions as appropriate. This alternative would include research and development and pilot programs to support current management and future decisions on spent fuel disposition.

3.2.5.2 SRS Options 4a, 4b, and 4c (Regionalization A). DOE analyzed three options for the regionalization of fuels by fuel type: dry storage (Option 4a), wet storage (Option 4b) and processing of existing SRS aluminum-clad fuels and storing the remaining fuel (Option 4c). This subalternative assumes that the SRS would manage:

- Its current inventory of 184.4 MTHM (203.3 tons) of aluminum-clad fuels, plus

- Approximately 28.7 MTHM (31.6 tons) of research reactor aluminum-clad fuel from other sites

The SRS would ship to the Idaho National Engineering Laboratory approximately:

- 5.4 MTHM (6.0 tons) of stainless steel-clad fuel
- 4.6 MTHM (5.1 tons) of commercial nonaluminum-clad fuel
- 11.9 MTHM (13.1 tons) of Zircaloy-clad spent fuel

DOE would manage a total of about 213.1 MTHM (234.9 tons) of spent nuclear fuel at the SRS under this subalternative. The site would receive shipments from other locations as existing space became available and as it shipped the nonaluminum-clad fuel.

3.2.5.2.1 Option 4a - Dry Storage — The actions that the SRS would undertake under this option, and the new facilities to be constructed, would be the same as for those described for Option 2a - Dry Storage under Alternative 2 (Decentralization) in Section 3.2.3.2.1.

This option would require an extensive research and development program into capabilities to examine, characterize, and can the SRS aluminum-clad fuel for dry storage.

3.2.5.2.2 Option 4b - Wet Storage — The SRS would carry out the same actions and construct the same types of facilities under this option as it would for Option 2b - Wet Storage under Alternative 2 (Decentralization) as described in Section 3.2.3.2.2. Research and development activities would also be similar to those conducted under this Decentralization alternative, except the SRS would not perform studies on nonaluminum-clad fuels.

3.2.5.2.3 Option 4c - Processing and Storage — Under this option, the SRS would process the existing aluminum-clad fuel as described for Option 2c - under Alternative 2 (Decentralization) and place the aluminum-clad fuel received from offsite into wet storage. The requirements for new construction would be different than in Option 2c, in that dry storage facilities would not be required because the nonaluminum-clad fuels would be shipped off the site. The small amount of aluminum-clad fuel to be received could be more readily stored in pools rather than developing new dry storage. Therefore, Option 4c would require DOE to construct a new fuel receiving, wet canning and wet storage facility to manage the fuel received after the major processing operations are completed. These facilities would be much smaller than those required for other alternatives.

3.2.5.3 SRS Options 4d, 4e, 4f, and 4g (Regionalization B). DOE analyzed the same three options for the regionalization of spent fuel on the basis of geographic location as for the other alternatives: dry storage (Option 4d), wet storage (Option 4e), and processing of existing aluminum-clad fuel and storing the remaining fuel (Option 4f). In addition, it assessed the option of shipping all SRS inventory offsite (Option 4g).

The amount of material that the SRS would manage if all the spent fuel in the East were shipped to the Site would total about 256.6 MTHM (282.9 tons). This would include the current SRS inventory of about 206.3 MTHM (227.4 tons) as detailed in Section 3.2.2 plus:

- 19.9 MTHM (21.9 tons) of aluminum-clad fuel
- 26.7 MTHM (29.4 tons) of commercial nonaluminum-clad fuel
- 1.0 MTHM (1.1 ton) of stainless steel-clad fuel
- 1.3 MTHM (1.4 tons) of experimental Zircaloy-clad fuel
- 1.4 MTHM (1.5 tons) of other experimental fuel

The activities that DOE would have to undertake at the SRS, and the facilities that it would have to build, under the dry storage, wet storage, or processing options would be very similar to those required for the Decentralization alternative (Section 3.2.3). The difference would be that the size of the storage facilities would be somewhat greater because the amount of fuel to be managed would be larger [256.6 MTHM (282.9 tons) versus 219.9 MTHM (242.4 tons)]. In addition, DOE would conduct additional research and development on the other fuel types that SRS would manage under these options.

3.2.5.3.1 Option 4d - Dry Storage — The actions that the SRS would undertake under this option, and the new facilities to be constructed, would be similar to those described for Option 2a - Dry Storage under Alternative 2 (Decentralization) in Section 3.2.3.2.1. This option would require an extensive research and development program into capabilities to examine, characterize, and can the SRS aluminum-clad fuel for dry storage.

3.2.5.3.2 Option 4e - Wet Storage — The SRS would carry out the same actions and construct the same types of facilities under this option as it would for Option 2b - Wet Storage under Alternative 2 (Decentralization) as described in Section 3.2.3.2.2. Research and development activities would also be similar to those conducted under this Decentralization alternative.

3.2.5.3.3 Option 4f - Processing and Storage — Under this option, the SRS would process the existing aluminum-clad fuel and place nonaluminum-clad fuel and aluminum-clad fuel received from offsite in dry storage as described for Option 2c - Processing with storage under Alternative 2 (Decentralization). The requirements for new facilities and for research and development would also be similar.

3.2.5.3.4 Option 4g - Shipment Off the Site – Under this option, the SRS would ship its current inventory of about 206.3 MTHM (227.4 tons) to the Oak Ridge Reservation. The activities and facilities required for this option are the same as those described below for Option 5d of the Centralization alternative (Section 3.2.6.2.4).

3.2.6 Alternative 5 - Centralization

3.2.6.1 Overview. Under this alternative, DOE would collect all current and future spent nuclear fuel inventories from DOE sites, the Navy, and other sources at a single location for management until final disposition. DOE would construct new facilities at the centralized site to accommodate the increased inventories. The originating sites would characterize and stabilize their spent nuclear fuel before shipping. They would then close their spent fuel facilities. This alternative would include the centralization of activities related to the treatment of spent nuclear fuel, including research and development and pilot programs to support future decisions on its disposition.

3.2.6.2 SRS Options 5a, 5b, 5c, and 5d. DOE analyzed four options for this alternative. Three deal with shipping all DOE spent nuclear fuel to the SRS for disposition and management in dry storage (Option 5a), wet storage (Option 5b), or by processing existing aluminum-clad fuel and storing the remaining fuel (Option 5c). The fourth case involves the shipment of all SRS fuel off the Site to another location (Option 5d). Options 5a, 5b, and 5c concern the following fuels:

- 65.2 MTHM (71.7 tons) of naval fuel
- 213.1 MTHM (234.9 tons) of aluminum-clad fuel
- 2103.2 MTHM (2,318.4 tons) of Hanford defense fuel
- 27.6 MTHM (30.4 tons) of graphite fuel
- 156.5 MTHM (172.5 tons) of commercial nonaluminum-clad fuel
- 96.5 MTHM (106.4 tons) of experimental stainless steel-clad fuel
- 78.0 MTHM (86.0 tons) of Zircaloy-clad fuel
- 1.7 MTHM (1.9 tons) of other fuel types

DOE would manage a total of about 2,741.8 MTHM (3,022.3 tons) of spent nuclear fuel at the SRS under the first three options. Options 5a and 5b would involve storing all the fuel on the Site. Option 5c would require processing the existing aluminum-clad fuel [184.4 MTHM (203.3 tons)] and placing the remaining nonaluminum-clad SRS fuels and all fuel received from other locations

[2,557.4 MTHM (2,819.0 tons)] into dry storage. The SRS could accept shipments from offsite sources and place them in storage as it built new facilities and transferred the onsite inventory.

Under Option 5d, shipments leaving the Site would amount to about 206.3 MTHM (227.4 tons), which is equal to the inventory of spent nuclear fuel at the SRS under Alternative 1.

3.2.6.2.1 Option 5a - Dry Storage — The actions that the SRS would undertake under this option would be the same as those described for Option 2a - Dry Storage under Alternative 2 (Decentralization) in Section 3.2.3.2.1. However, the number and size of the new facilities needed to implement this centralization option would be much greater because of the larger volume of fuel that the Site would manage. In addition, DOE would have to build a new Expended Core Facility at the SRS to examine and characterize the naval fuels.

This option would require an extensive research and development program into capabilities to examine, characterize, and can SRS and other fuel types before their placement in a dry storage vault. DOE would also carry out research and development into other aspects of the management of the spent fuels, including those related to its ultimate disposition.

3.2.6.2.2 Option 5b - Wet Storage — Under this option, DOE would undertake actions similar to those described in Section 3.2.3.2.2 for Option 2b - Wet Storage under Alternative 2. As with Option 5a (Dry Storage), the SRS would have to build major new facilities to manage the large volume of fuel it would receive. DOE would also have to build a new Expended Core Facility at the SRS. Research and development would be greatly expanded as well.

3.2.6.2.3 Option 5c - Processing and Storage — DOE would process the current inventory of aluminum-clad spent fuel under this option in the same manner as described for the other alternatives. All other fuel onsite and all fuel received from elsewhere would be canned and placed in new dry storage facilities. The SRS would shut down the F- and H-Area separations facilities after processing the existing inventory of aluminum-clad fuel. Thereafter, any aluminum-clad fuel sent to the SRS would be placed in dry storage.

This option would require major new facilities, including a new Expended Core Facility. DOE would also conduct extensive research and development in spent fuel management.

3.2.6.2.4 Option 5d - Shipment Off the Site — DOE would consolidate and prepare all spent nuclear fuel on the SRS for shipment to another DOE site; this would require the construction of a new fuel characterization facility. Some fuels could require canning before shipment. SRS would use existing facilities to accomplish this. DOE would then close all SRS spent nuclear fuel-related facilities.

DOE would conduct research and development into methods of stabilizing, canning, and transporting aluminum-clad fuels, particularly that which is corroded or otherwise degraded.

3.3 Comparison of Alternatives

Table 3-3 summarizes the environmental consequences of the five alternatives. Chapter 5 presents detailed descriptions of these consequences.

In general, the levels of impacts associated with Alternatives 1 through 4 would be similar because the amounts of spent nuclear fuel that DOE would manage at the SRS under these cases would be approximately the same [e.g., about 206 to 257 MTHM (227 to 283 tons)] and activities would extend throughout the full 40-year management period. The lowest level of impact at SRS would occur under Option 4g or Option 5d (Regionalization or Centralization at another site) because DOE would ship the SRS spent fuel off the Site well before the management period ended in 2035. Alternative 5, under which DOE would ship all spent nuclear fuel to the SRS, would result in the greatest onsite impacts; the Site would have to manage approximately 2,741.8 MTHM (3,022.3 tons) of spent fuel.

Table 3-3. Comparison of impacts for the five alternatives.

ALTERNATIVE 1 - NO ACTION

	Option 1 Wet Storage
Land Use	No new facilities would be required.
Socioeconomics	No new operations jobs and only about 50 construction jobs would be created.
Cultural Resources	No new construction would be carried out. No impacts are anticipated.
Aesthetics and Scenic Resources	Facilities are in an existing industrial area not visible from public access roads or from off the Site. No impacts are anticipated. Emissions would not impact visibility.
Geology	No minerals of economic value are in affected area. No impacts are anticipated.
Air Resources	Emissions of criteria air pollutants and toxic air pollutants would be only a small fraction of air quality standards.
Water Resources	This option would not require use of additional surface water beyond the 75.7 billion liters (20 billion gallons) per year that the SRS withdraws at present. This option would not require withdrawals of additional groundwater beyond the 14.0 billion liters (3.7 billion gallons) per year the SRS uses. Activities related to this option currently use about 35.1 million liters (9.3 million gallons) of groundwater per year. Impacts would be minimal. No perennial streams or other surface waters would be affected. Accidental releases could contaminate shallow groundwater that is not a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.
Ecological Resources	Minor disturbance of wildlife due to traffic would occur. No wetlands or threatened or endangered species would be affected.
Noise	The only noise experienced by offsite populations would be generated by employee traffic and by truck and rail deliveries. There would be no change in traffic noise impacts.
Traffic and Transportation	This option would not increase site traffic. Number of LCF[f], normal transport: Worker: 6.0×10^{-4} Public: 7.0×10^{-5}
Occupational and Public Health and Safety (Radiological)	Maximum LCF[f] probabilities: Worker: 4×10^{-5} Offsite population: 4×10^{-14} (air) 1×10^{-14} (water) Annual LCF[f] incidences: Worker: 8×10^{-5} Offsite population: 2×10^{-9}

Table 3-3. (continued).

	Option 1 Wet Storage
Occupational and Public Health and Safety (Nonradiological)	Hazard index: Worker: 2×10^{-6} Maximally exposed individual: 2×10^{-7}
Utilities and Energy	Minimal changes in demand for electricity, steam, domestic water and wastewater treatment would occur. Current SRS capacities are adequate for these additions. Impacts would be minimal.
Materials and Waste Management	Annual average volume of waste generated (cubic meters)[b]: LLW: 400 TRU: 17 HLW: 0.4 No impact on site waste management capacities.
Accidents[c]	Greatest point estimate of risk[d]: Worker: Data not calculated[e] Colocated worker: 7.7×10^{-7} Maximally exposed individual: 1.6×10^{-7} Offsite population: 1.4×10^{-3}

a. Not applicable.
b. LLW = low-level waste; TRU = transuranic waste; HLW = high-level waste.
c. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.
d. Units for adjusted point estimates of risk are given in tenns of potential fatal cancers per year.
e. The safety analysis reports from which infonnation was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.
f. LCF = latent cancer fatalities.

Table 3-3. (continued).

ALTERNATIVE 2 - DECENTRALIZATION

	Option 2a Dry Storage	Option 2b Wet Storage	Option 2c Processing
Land Use	Most new construction would be in parts of F- and H-Areas already dedicated to industrial use. Impacts would be minimal.	Same as Option 2a.	Same as Option 2a.
Socioeconomics	Operations jobs would be filled by current employees. A maximum of about 600 construction jobs would be created.	Same as Option 2a.	Operations jobs would be filled by current employees. A maximum of about 550 construction jobs would be created.
Cultural Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Aesthetics and Scenic Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Geology	Same as Option 1.	Same as Option 1.	Same as Option 1.
Air Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Water Resources	New withdrawals of approximately 6.1 million liters (1.6 million gallons) per year of cooling water from Savannah River would be required. Impacts would be minimal.	New withdrawals of approximately 7.2 million liters (1.9 million gallons) per year of cooling water from Savannah River would be required. Impacts would be minimal.	New withdrawals of approximately 311 million liters (82.2 million gallons) per year of cooling water from Savannah River would be required. Impacts would be minimal.
	Additional groundwater withdrawals would total about 48.7 million liters (12.9 million gallons) per year. Impacts would be minimal.	Additional groundwater withdrawals would total about 50.6 million liters (13.4 million gallons) per year. Impacts would be minimal.	Same as Option 2a.
	No perennial streams or other surface waters would be affected.	No perennial streams or other surface waters would be affected.	No perennial streams or other surface waters would be affected.
	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.

Table 3-3. (continued).

	Option 2a Dry Storage	Option 2b Wet Storage	Option 2c Processing
Ecological Resources	Small increase in traffic would cause slight increase in road kills and in disturbance of wildlife due to noise. Impacts would be minimal.	Same as Option 2a.	Small increases in traffic would cause small increase in road kills and in disturbance of wildlife due to noise. Impacts would be minimal.
	No wetlands or threatened or endangered species would be affected.	Same as Option 2a.	Same as Option 2a.
Noise	Only noise experienced by communities would be generated by employee traffic and by truck and rail deliveries.	Same as Option 2a.	Same as Option 2a.
	Changes in traffic levels are expected to result in only very small changes in noise impacts.		
Traffic and Transportation	This option would increase site traffic slightly.	Same as Option 2a.	This option would increase site traffic slightly.
	Number of LCF[a], normal transport: Worker: 1.0×10^{-3} Public: 1.2×10^{-4}		Number of LCF[a], normal transport: Worker: 2.1×10^{-4} Public: 1.9×10^{-5}
Occupational and Public Health and Safety (Radiological)	Maximum LCF[a] probabilities: Worker: 3×10^{-5} Offsite population: 4×10^{-14} (air) 1×10^{-14} (water)	Maximum LCF[a] probabilities: Worker: 4×10^{-5} Offsite population: 5×10^{-14} (air) 2×10^{14} (water)	Maximum LCF[a] probabilities: Worker: 6×10^{-5} Offsite population: 2×10^{-7} (air) 6×10^{-8} (water)
	Annual LCF[a] incidences: Worker: 7×10^{-5} Offsite population: 2×10^{-9}	Annual LCF[a] incidences: Worker: 8×10^{-5} Offsite population: 2×10^{-9}	Annual LCF[a] incidences: Worker: 3×10^{-2} Offsite population: 8×10^{-3}
Occupational and Public Health and Safety (Nonradiological)	Same as Option 1.	Same as Option 1.	Hazard index: Worker: 6×10^{-3} Maximally exposed individual: 5×10^{-4}
Utilities and Energy	Requirements would increase 3 to 7 percent above present levels. Current SRS capacities are adequate for these increases.	Same as Option 2a.	Very similar to Option 2a.
Materials and Waste Management	Annual average volume of waste generated (cubic meters)[b]: LLW: 400 TRU: 18 HLW: 0.4	Same as Option 2a.	Annual average volume of waste generated (cubic meters)[b]: LLW: 800 TRU: 19 HLW: 2.3[c]
	No impact on site capacities.		No impact on site capacities.

Table 3-3. (continued).

	Option 2a Dry Storage	Option 2b Wet Storage	Option 2c Processing
Accidents[d]	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 1.6×10^{-6} Maximally exposed individual: 3.3×10^{-7} Offsite population: 2.8×10^{-3}	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 1.7×10^{-6} Maximally exposed individual: 3.5×10^{-7} Offsite population: 3.0×10^{-3}	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 7.7×10^{-7} Maximally exposed individual: 1.6×10^{-7} Offsite population: 1.4×10^{-3}

a. NA = not applicable.
b. LLW = low-level waste; TRU = transuranic waste; HLW = high-level waste.
c. High-level waste will be generated only during approximately the first 10 years.
d. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.
e. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
f. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.
g. LCF = latent cancer fatalities.

Table 3-3. (continued).

ALTERNATIVE 3 - 1992/1993 PLANNING BASIS

	Option 3a Dry Storage	Option 3b Wet Storage	Option 3c Processing
Land Use	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Socioeconomics	Same as Option 2a.	Operations jobs would be filled by current employees. A maximum of about 650 construction jobs would be created.	Same as Option 2c.
Cultural Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Aesthetics and Scenic Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Geology	Same as Option 1.	Same as Option 1.	Same as Option 1.
Air Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Water Resources	Same as Option 2a.	Same as Option 2b.	Same as Option 2c.
Ecological Resources	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Noise	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Traffic and Transportation	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Occupational and Public Health and Safety (Radiological)	Same as Option 2a.	Same as Option 2b.	Same as Option 2c.
Occupational and Public Health and Safety (Nonradiological)	Same as Option 1.	Same as Option 1.	Same as Option 2c.
Utilities and Energy	Same as Option 2a.	Same as Option 2a.	Very similar to Option 2a.
Materials and Waste Management	Same as Option 2a.	Same as Option 2a.	Annual average volume of waste generated (cubic meters)[a]: LLW: 750 TRU: 19 HLW: 1.7[b] No impact on site capacities.
Accidents[c]	Greatest point estimate of risk[d]: Worker: Data not calculated[e] Colocated worker: 1.9×10^{-6} Maximally exposed individual: 4.0×10^{-7} Offsite population: 3.4×10^{-3}	Same as Option 3a.	Greatest point estimate of risk[d]: Worker: Data not calculated[e] Colocated worker: 1.1×10^{-6} Maximally exposed individual: 2.3×10^{-7} Offsite population: 2.0×10^{-3}

a. LLW = low-level waste; TRU = transuranic waste; HLW = high-level waste.

b. High-level waste will be generated only during approximately the first 10 years.

c. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.

d. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.

e. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.

Table 3-3. (continued).

ALTERNATIVE 4 - REGIONALIZATION A (By Fuel Type)

	Option 4a Dry Storage	Option 4b Wet Storage	Option 4c Processing
Land Use	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Socioeconomics	Same as Option 3b.	Same as Option 3b.	Same as Option 2c.
Cultural Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Aesthetics and Scenic Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Geology	Same as Option 1.	Same as Option 1.	Same as Option 1.
Air Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Water Resources	Same as Option 2a.	Same as Option 2b.	Very similar to Option 2c.
Ecological Resources	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Noise	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Traffic and Transportation	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Occupational and Public Health and Safety (Radiological)	Same as Option 2a.	Same as Option 2b.	Maximum LCF[a] probabilities: Same as Option 2c. Annual LCF[a] incidences: Worker: 3×10^{-2} Offsite population: 9×10^{-3}
Occupational and Public Health and Safety (Nonradiological)	Same as Option 1.	Same as Option 1.	Same as Option 2c.
Utilities and Energy	Very similar to Option 2a.	Same as Option 2a.	Very similar to Option 2a.
Materials and Waste Management	Same as Option 1.	Same as Option 1.	Annual average volume of waste generated (cubic meters)[b]: LLW: 790 TRU: 18 HLW: 2.3[c] No impact on site capacities.

Table 3-3. (continued).

	Option 4a Dry Storage	Option 4b Wet Storage	Option 4c Processing
Accidents[d]	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 2.1×10^{-6} Maximally exposed individual: 4.4×10^{-7} Offsite population: 3.7×10^{-3}	Same as Option 3a.	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 1.3×10^{-6} Maximally exposed individual: 2.8×10^{-7} Offsite population: 2.4×10^{-3}

a. LCF = latent cancer fatalities.
b. LLW = low-level waste; TRU = transuranic waste; HLW = high-level waste.
c. High-level waste will be generated only during approximately the first 10 years.
d. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.
e. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
f. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.

Table 3-3. (continued).

ALTERNATIVE 4 - REGIONALIZATION B (By Location)[a]

	Option 4d Dry Storage	Option 4e Wet Storage	Option 4f Processing
Land Use	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Socioeconomics	Operations jobs would be filled by current employees. A maximum of about 700 construction jobs would be created.	Operations jobs would be filled by current employees. A maximum of about 800 construction jobs would be created.	Same as Option 3b.
Cultural Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Aesthetics and Scenic Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Geology	Same as Option 1.	Same as Option 1.	Same as Option 1.
Air Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Water Resources	Same as Option 2a.	Same as Option 2b.	Very similar to Option 2c.
Ecological Resources	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Traffic and Transportation	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Occupational and Public Health and Safety (Radiological)	Maximum LCF[c] probabilities: Worker: 4×10^{-5} Offsite population: 5×10^{-14} (air) 2×10^{-14} (water) Annual LCF[c] incidences: Worker: 8×10^{-5} Offsite population: 2×10^{-9}	Maximum LCF[c] probabilities: Worker: 5×10^{-5} Offsite population: 6×10^{-14} (air) 2×10^{-14} (water) Annual LCF[c] incidences: Worker: 1×10^{-4} Offsite population: 2×10^{-9}	Maximum LCF[c] probabilities: Worker: 7×10^{-5} Offsite population: 2×10^{-7} (air) 6×10^{-8} (water) Annual LCF[c] incidences: Worker: 3×10^{-2} Offsite population: 9×10^{-3}
Occupational and Public Health and Safety (Nonradiological)	Hazard index: Worker: 2×10^{-6} Maximally exposed individual: 3×10^{-7}	Same as Option 4d.	Hazard index: Worker: 8×10^{-3} Maximally exposed individual: 6×10^{-4}
Utilities and Energy	Same as Option 2a.	Very similar to Option 2a.	Very similar to Option 2a.
Materials and Waste Management	Same as Option 1.	Same as Option 1.	Same as Option 4c.

Table 3-3. (continued).

	Option 4d Dry Storage	Option 4e Wet Storage	Option 4f Processing
Accidents[b]	Greatest point estimate of risk[c]: Worker: Data not calculated[d] Colocated worker: 2.0×10^{-6} Maximally exposed individual: 4.1×10^{-7} Offsite population: 3.5×10^{-3}	Same as Option 4d	Greatest point estimate of risk[c]: Worker: Data not calculated[d] Colocated worker: 1.2×10^{-6} Maximally exposed individual: 2.5×10^{-7} Offsite population: 2.1×10^{-3}

a. Impacts for Option 4g, Ship Offsite, would be the same as for Option 5d as described in the last entry in this table.

b. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.

c. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.

d. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.

e. LCF = latent cancer fatalities.

Table 3-3. (continued).

ALTERNATIVE 5 - CENTRALIZATION

	Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing
Land Use	Most new construction would be in parts of F- and H-Areas already dedicated to industrial use. Additional maximum of 0.4 square kilometer (100 acres) would be converted from pine plantation to industrial use. Impacts would be minimal.	Same as Option 5a.	Same as Option 5a.
Socioeconomics	Operations jobs would be filled by present employees. A maximum of about 2,550 construction jobs would be created.	Operations jobs would be filled by present employees. A maximum of about 2,700 construction jobs would be created.	Operations jobs would be filled by present employees. A maximum of about 2,550 construction jobs would be created.
Cultural Resources	No known historical, archeological, or paleontological resources are in areas to be affected. All areas are classified as having low or moderate probability of containing archeological site. Impact is unlikely.	Same as Option 5a.	Same as Option 5a.
Aesthetics and Scenic Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Geology	Same as Option 1.	Same as Option 1.	Same as Option 1.
Air Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Water Resources	Same as Option 2a.	Same as Option 2b.	Same as Option 2c.
	Additional groundwater withdrawals would total about 67.7 million liters (17.9 million gallons) per year. Impacts would be minimal.	Additional groundwater withdrawals would total about 69.6 million liters (18.4 million gallons) per year. Impacts would be minimal.	Same as Option 5a.
	No perennial streams or other surface waters would be affected.	Same as Option 5a.	Same as Option 5a.
	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.

Table 3-3. (continued).

	Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing
Ecological Resources	Same as Option 2a, plus Loss of up to 0.4 square kilometer (100 acres) of loblolly pine. Impacts would be minor.	Same as Option 5a.	Same as Option 5a, plus Increased disturbance due to more worker traffic. Impacts would be minor.
Noise	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Traffic and Transportation	Same as Option 2a.	This option would increase site traffic by about 17 percent. Impacts would be small. Number of LCFs[g] would be same as for Option 2b for normal transport.	Same as Option 2c.
Occupational and Public Health and Safety (Radiological)	Maximum LCF[g] probabilities: Worker: 4×10^{-4} Offsite population: 5×10^{-13} (air) 2×10^{-13} (water) Annual LCF[g] incidences: Worker: 9×10^{-4} Offsite population: 2×10^{-8}	Maximum LCF[g] probabilities: Worker: 5×10^{-4} Offsite population: 6×10^{-13} (air) 2×10^{-13} (water) Annual LCF[g] incidences: Worker: 1×10^{-3} Offsite population: 3×10^{-8}	Maximum LCF[g] probabilities: Worker: 6×10^{-4} Offsite population: 2×10^{-7} (air) 6×10^{-8} (water) Annual LCF[g] incidences: Worker: 3×10^{-2} Offsite population: 9×10^{-3}
Occupational and Public Health and Safety (Nonradiological)	Same as Option 1.	Same as Option 1.	Same as Option 2c.
Utilities and Energy	Similar to Option 2a.	Similar to Option 2a.	Requirements for electricity would increase by about 17 percent. Other increases would be similar to Option 2c. Impacts would be minor.
Materials and Waste Management	Annual average volume of waste generated (cubic meters)[b]: LLW: 400 TRU: 16 HLW: 0 No impact on site capacities.	Annual average volume of waste generated (cubic meters)[b]: LLW: 400 TRU: 20 HLW: 2.3[c] No impact on site capacities.	Annual average volume of waste generated (cubic meters)[b]: LLW: 800 TRU: 20 HLW: 2.3[c] No impact on site capacities.

Table 3-3. (continued).

	Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing
Accidents[d]	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 4.0×10^{-6} Maximally exposed individual: 8.4×10^{-7} Offsite population: 7.2×10^{-3}	Same as Option 5a.	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 3.3×10^{-6} Maximally exposed individual: 6.8×10^{-7} Offsite population: 5.8×10^{-3}

a. NA = not applicable.
b. LLW = low-level waste; TRU = transuranic waste; HLW = high-level waste.
c. High-level waste will be generated only during approximately the first 10 years.
d. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.
e. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
f. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.
g. LCF = latent cancer fatalities.

Table 3-3. (continued).

ALTERNATIVE 5 - CENTRALIZATION
ALTERNATIVE 4 - REGIONALIZATION B

	Option 4g and Option 5d[b] Ship Out
Land Use	Same as Option 1.
Socioeconomics	No new operations jobs and only about 200 construction jobs would be created.
Cultural Resources	Same as Option 1.
Aesthetics and Scenic Resources	Same as Option 1.
Geology	Same as Option 1.
Air Resources	Same as Option 1.
Water Resources	This option would require new withdrawals of approximately 3.0 million liters (790 thousand gallons) per year of cooling water from the Savannah River. Impacts would be minimal. It also would require additional groundwater withdrawals of about 38.1 million liters (10.1 million gallons) per year. Impacts would be minimal. Impacts to surface water and groundwater would be similar to those from Option 1.
Ecological Resources	Same as Option 1.
Noise	Same as Option 2a.
Traffic and Transportation	NA[a]
Occupational and Public Health and Safety (Radiological)	Less than Option 1.
Occupational and Public Health and Safety (Nonradiological)	Same as Option 1.
Utilities and Energy	Requirements would increase 2 to 6 percent above current levels during first 10 years. Current SRS capacities are adequate for these increases.
Materials and Waste Management	Annual average volume of waste generated initial 10 years only (cubic meters)[c]: LLW: 400 TRU: 18 HLW: 0

Table 3-3. (continued).

	Option 4g and Option 5d[b] Ship Out
Accidents[d]	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated Worker: Option 4g: 8.1×10^{-7} Option 5d: 8.2×10^{-7} Maximally exposed individual: Option 4g: 1.7×10^{-7} Option 5d: 1.7×10^{-7} Offsite population: Option 4g: 1.4×10^{-3} Option 5d: 1.4×10^{-3}

a. NA = not applicable.

b. Impacts for Option 4g (Regionalization-B) are the same as for Option 5d.

c. LLW = low-level waste; TRU = transuranic waste; HLW = high-level waste.

d. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.

e. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.

f. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.

4. AFFECTED ENVIRONMENT

4.1 Overview

This section describes the existing environment at the Savannah River Site (SRS) and nearby areas. Its purpose is to support the assessment of environmental consequences of the alternative actions regarding spent nuclear fuels described in Chapter 3. Chapter 5 describes the environmental consequences in detail.

4.2 Land Use

The SRS occupies an area of approximately 198,000 acres (800 square kilometers) in western South Carolina, in a generally rural area about 25 miles (40 kilometers) southeast of Augusta, Georgia. The SRS, which is bordered by the Savannah River to the southwest, includes portions of Aiken, Barnwell, and Allendale Counties (Figure 2-1).

Land use on the SRS falls into three major categories: forest/undeveloped, water/wetlands, and developed facilities. About 181,500 acres (735 square kilometers) of the SRS area are undeveloped (USDA 1991a). Approximately 90 percent of this undeveloped area is forested (Cummins et al. 1991). In 1952, an interagency agreement between the U.S. Department of Energy [DOE, which was then the Atomic Energy Commission (AEC)] and the Forest Service, U.S. Department of Agriculture, created an SRS forest management program. In 1972, the AEC designated the SRS as a National Environmental Research Park (NERP); at present, approximately 14,000 acres (57 square kilometers or 7 percent) of the SRS area are designated as "Set-Asides," areas specifically protected for environmental research activities that are coordinated either through the University of Georgia Savannah River Ecology Laboratory (SREL) or the Savannah River Technology Center (SRTC; Davis 1994). Administrative, production, and support facilities occupy approximately 5 percent of the total SRS land area.

DOE is considering decisions that could affect the long-range land use of the SRS. Programmatic decisions on the reconfiguration of the nuclear weapons complex, spent nuclear fuel interim strategies, and waste management and environmental restoration activities that could result in significant changes in the SRS mission are in the early stages of discussion. In the shorter term, however, a Land Use Technical Committee consisting of representatives from DOE, Westinghouse

Savannah River Company, and various stakeholder groups is evaluating alternative land use strategies and potential future uses. These activities are consistent with the guidelines for land use plans contained in DOE Order 4320.1B, "Site Development Planning," and in the Resource Conservation and Recovery Act (RCRA) and the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA).

Land use bordering SRS is primarily forest and agricultural. There is also a significant amount of open water and nonforested wetlands along the Savannah River valley. Incorporated and industrial areas are the only other significant use of land in the vicinity (Figure 4-1). None of the three counties in which the SRS is located has zoned any of the Site land. The only adjacent area with any zoning is the Town of New Ellenton, which has two zoning categories for lands that bound SRS - urban development and residential development. The closest residences to the SRS boundary include several within 200 feet (61 meters) of the Site perimeter to the west, north, and northeast.

Various industrial, manufacturing, medical, and farming operations are conducted in areas surrounding the Site. Major industrial and manufacturing facilities in the area include textile mills, plants producing polystyrene foam and paper products, chemical processing plants, and a commercial nuclear power plant. Farming is diversified in the region and includes crops such as peaches, watermelon, cotton, soybeans, corn, and small grains.

There is a wide variety of public outdoor recreation facilities in the SRS region (Figure 4-2). Federal outdoor recreation facilities include portions of the Sumter National Forest [47 miles (75 kilometers) to the northwest of the Site], the Santee National Wildlife Refuge [50 miles (80 kilometers) to the east], and the Clarks Hill/Strom Thurmond Reservoir, a U.S. Army Corps of Engineers impoundment [43 miles (70 kilometers) to the northwest]. There are also a number of state, county, and local parks in the region, most notably Redcliffe Plantation, Rivers Bridge, Barnwell and Aiken County State Parks in South Carolina, and Mistletoe State Park in Georgia (HNUS 1992a).

The SRS is a controlled area with public access limited to through traffic on South Carolina Highway 125 (SRS Road A), U.S. Highway 278, SRS Road 1, and the CSX railway. The SRS does not contain any public recreation facilities. However, the SRS conducts controlled deer hunts each fall, from mid-October through mid-December; hunters can also kill feral hogs during these hunts.

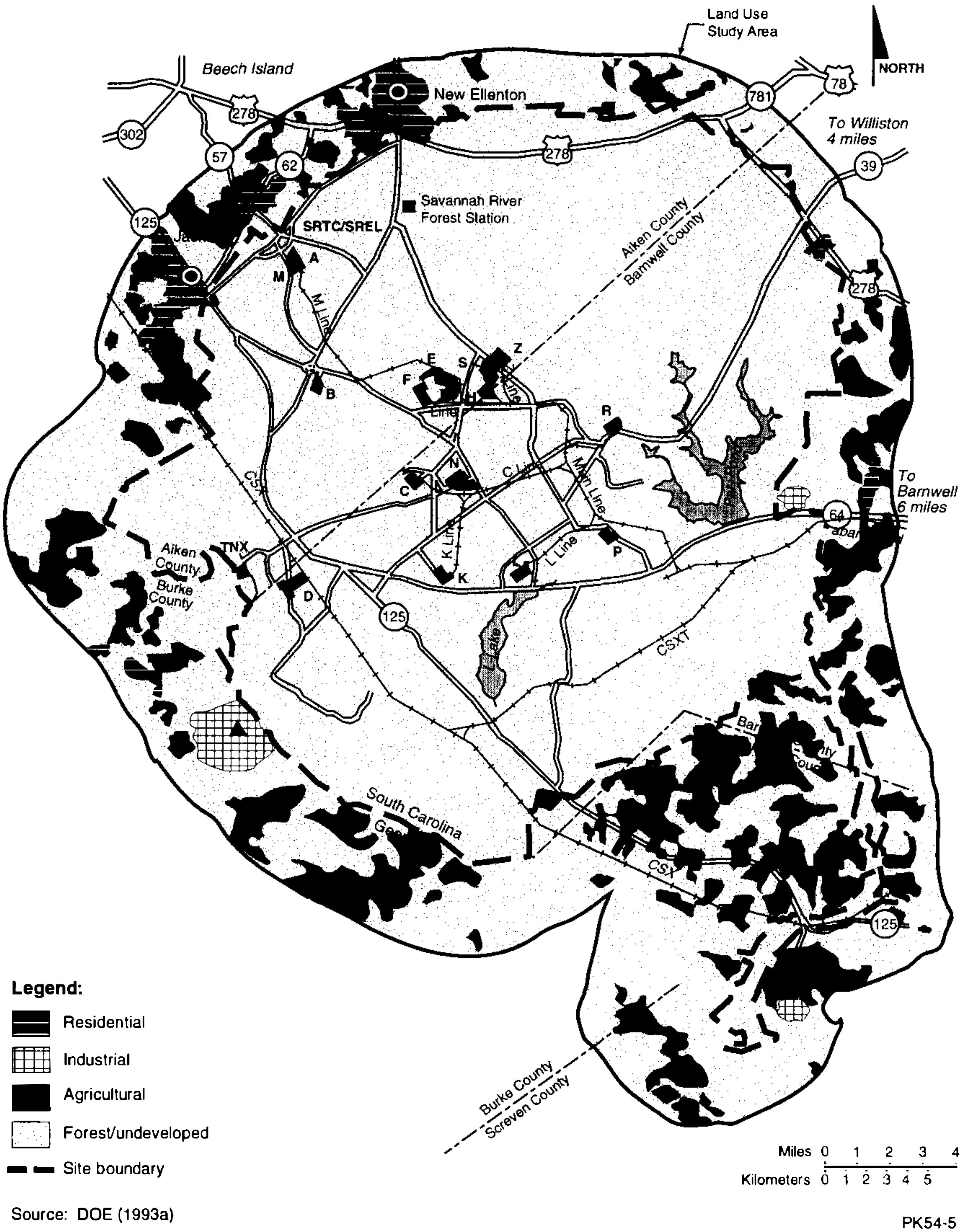
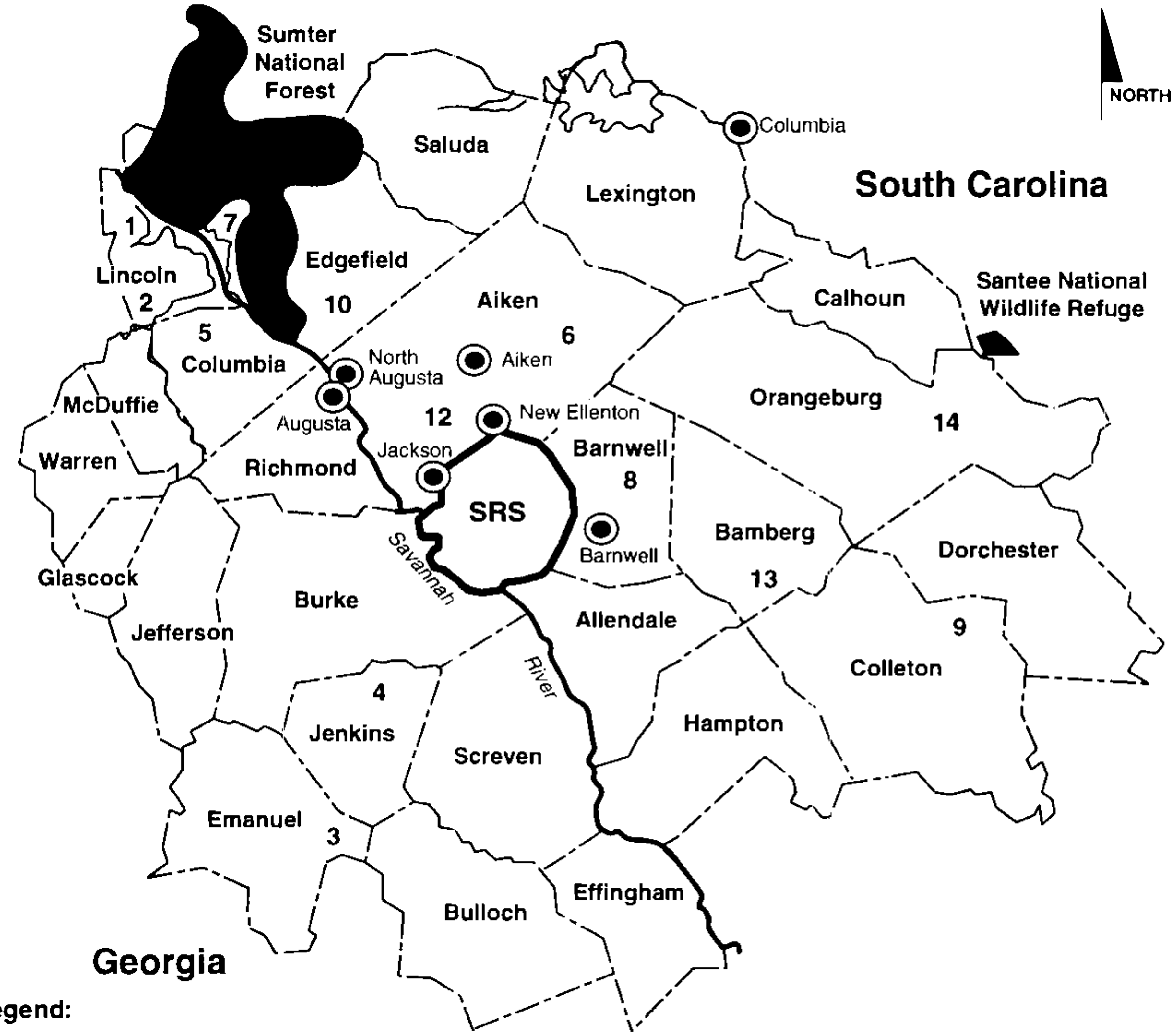


Figure 4-1. Generalized land use at the Savannah River Site and vicinity.



Legend:

Georgia

1 Clarks Hill Reservoir
2 Elijah Clark State Park
3 George L. Smith State Park
4 Magnolia Springs State Park
5 Mistletoe State Park

South Carolina

6 Aiken State Park
7 Baker Creek State Park
8 Barnwell State Park
9 Colleton State Park
10 Hamilton Branch State Park
11 Hickory Knob State Park
12 Redcliffe Plantation State Park
13 Rivers Bridge State Park
14 Santee State Park

Source: HNUS (1992a)

Miles 0 10 20 30 40
Kilometers 0 10 20 30 40 50 60

PK54-6

Figure 4-2. Federal and state forests and parks within a 2-hour drive from Savannah River Site.

The intent of the hunts is to control the resident populations of these animals and to reduce animal-vehicle accidents on SRS roads.

No onsite areas are subject to Native American treaty rights. The SRS does not contain any prime farmland.

4.3 Socioeconomics

This section discusses baseline socioeconomic conditions within a region of influence where approximately 90 percent of the SRS workforce lived in 1992. The SRS region of influence includes Aiken, Allendale, Bamberg, and Barnwell Counties in South Carolina, and Columbia and Richmond Counties in Georgia (Figure 4-2).

4.3.1 Employment and Labor Force

The labor force living in the region of influence increased from about 150,550 to 209,000 between 1980 and 1990. In 1990, approximately 75 percent of the total labor force in the region of influence lived in Richmond and Aiken Counties. Assuming a constant unemployment rate of 5.8 percent, the regional labor force is likely to increase to approximately 257,000 by 1995 (Table 4-1).

Between 1980 and 1990, total employment in the region of influence increased from 139,504 to 199,161, an average annual growth rate of approximately 5 percent. Table 4-1 lists projected employment data for the six-county region of influence. As shown, by 1995 employment levels should increase 22 percent to approximately 242,000. The unemployment rates for 1980 and 1990 were 7.3 percent and 4.7 percent, respectively (HNUS 1992a).

In 1990, employment at the SRS was 20,230 (DOE 1993a), representing 10 percent of the employment in the region of influence. In Fiscal Year 1992, employment at the SRS increased approximately 15 percent to 23,351, with an associated payroll of more than $1.1 billion. Due to planned budget reductions, Site employment could decline by as many as 4,200 jobs (Fiori 1995). As shown in Table 4-1, this would reduce Site employment to approximately 15,800 by 1996.

Table 4-1. Forecast employment and population data for the Savannah River Site and the region of influence.[a]

Year	Labor Force (Region)	Employment (Region)	SRS Employment[b]	Population (Region)
1994	254,549	239,785	21,500	456,892
1995	256,935	242,033	20,000	461,705
1996	258,500	243,507	15,800	465,563
1997	260,680	245,561	15,800	468,665
1998	263,121	247,860	15,800	471,176
1999	265,694	250,284	15,800	473,186
2000	268,430	252,861	15,800	474,820
2001	271,265	255,532	15,800	476,179
2002	274,238	258,332	15,800	477,332
2003	277,318	261,234	15,800	478,340
2004	280,415	264,151	15,800	479,182

a. Source: HNUS (1993).
b. Sources: Turner (1994), Fiori (1995).

4.3.2 Personal Income

Personal income in the six-county region has doubled during the past two decades, increasing from approximately $3.4 billion in 1970 to almost $6.9 billion by 1989 (in constant 1991 dollars). Together, Richmond and Aiken Counties accounted for 75.4 percent of the personal income in the region of influence in 1989, because these two counties provide most of the employment opportunities in the region. Personal income in the region is likely to increase 3 percent to approximately $7.1 billion by 1995 and to almost $8.2 billion by 2000 (HNUS 1992a).

4.3.3 Population

Between 1980 and 1990, the population in the region of influence increased 13 percent from 376,058 to 425,607. More than 88 percent of the 1990 population lived in Aiken (28.4 percent), Columbia (15.5 percent), and Richmond (44.6 percent) Counties. Table 4-1 also lists population data for the region of influence forecast to 2004. According to census data, in 1990 the estimated average number of persons per household in the six-county region was 2.72, and the median age of the population was 31.2 years (HNUS 1992a).

4.3.4 Housing

From 1980 to 1990, the number of year-round housing units in the six-county region increased 23.2 percent from 135,866 to 167,356. In 1990, approximately 68 percent of the total housing units were single-family units, 18 percent were multifamily units, and 14 percent were mobile homes. In the same year, the region had a 4.7-percent vacancy rate with 7,818 available unoccupied housing units. Of the available unoccupied units, 29 percent (2,267) were available for sale and 71 percent (5,551) were available for rent (HNUS 1992a).

4.3.5 Community Infrastructure and Services

Public education facilities in the six-county region include 95 elementary and intermediate schools and 25 high schools. Aside from the public school systems, 42 private schools and 16 post-secondary facilities are available to residents in the region (HNUS 1992a).

Based on a combined average daily attendance for elementary and high school students in the region of influence in 1988, the average number of students per teacher was 16. The highest ratio was in Columbia County high schools where there were 19 students per teacher (1987-1988). The lowest ratio occurred in Barnwell County's District 29 high school, which had only 12 students per teacher (1988-1989) (HNUS 1992a).

The six-county region has 14 major public sewage treatment facilities with a combined design capacity of 302.2 million liters (79.8 million gallons) per day. In 1989, these systems were operating at approximately 56 percent of capacity, with an average daily flow of 170 million liters (44.9 million gallons) per day. Capacity utilization ranged from 45 percent in Aiken County to 80 percent in Barnwell County (HNUS 1992a).

There are approximately 120 public water systems in the region of influence. About 40 of these county and municipal systems are major facilities, while the remainder serve individual subdivisions, water districts, trailer parks, and miscellaneous facilities. In 1989, the 40 major facilities had a combined total capacity of 576.3 million liters (152.2 million gallons) per day. With an average daily flow rate of approximately 268.8 million liters (71 million gallons) per day, these systems were operating at 47 percent of total capacity in 1989. Facility utilization rates ranged from 13 percent in Allendale County to 84 percent in the City of Aiken (HNUS 1992a).

Eight general hospitals operate in the six-county region with a combined bed capacity in 1987 of 2,433 (5.7 beds per 1,000 population). Four of the eight general hospitals are in Richmond County; Aiken, Allendale, Bamberg, and Barnwell Counties each have one general hospital. Columbia County has no hospital. In 1989, there were approximately 1,295 physicians serving the regional population, which represents a physician-to-population ratio of 3 to 1,000. This ratio ranged from 0.8 physician per 1,000 people in Aiken and Allendale Counties to 5.4 physicians per 1,000 people in Richmond County (HNUS 1992a).

Fifty-six fire departments provide fire protection services in the region of influence. Twenty-seven of these are classified as municipal fire departments, but many provide protection to rural areas outside municipal limits. The average number of firefighters in the region in 1988 was 3.8 per 1,000 people, ranging from 1.6 per 1,000 in Richmond County to 10.2 per 1,000 in Barnwell County (HNUS 1992a).

The county sheriff departments and municipal police departments provide most law enforcement services in the region of influence. In addition, state law enforcement agents and state troopers assigned to each county provide protection and assist county and municipal law enforcement officers. In 1988, the average ratio in the region of full-time police officers employed by state, county, and local agencies per 1,000 population was 2.0. This ratio ranged from 1.4 per 1,000 in Columbia County to 2.5 per 1,000 in Richmond County (HNUS 1992a).

4.3.6 Government Fiscal Structure

This section discusses the fiscal structure of Aiken and Barnwell Counties because these two counties would have the greatest potential for fiscal impacts from changes at SRS.

Public services provided by Aiken County are funded principally through the county's general fund. In Fiscal Year 1988, revenues and expenditures of this fund were $15.5 million and $18 million, respectively. The current property tax rate is 55.8 mills for county operations and 8.0 mills for debt service. Long-term general obligation bond indebtedness was $9.3 million at the end of Fiscal Year 1988, and reserve general obligation bond indebtedness was $5.5 million. The assessed value of property in the county was $182.5 million in Fiscal Year 1988 (HNUS 1992a).

Assuming revenues and expenditures increase in proportion to projected growth in the employment and population, estimated revenues and expenditures for Aiken County over the period from Fiscal Year 1990 to Fiscal Year 2000 will be $15.6 million to $17.0 million (in constant 1988 dollars) (HNUS 1992a).

Public services provided by Barnwell County also are funded principally through the county's general fund. In Fiscal Year 1988, revenues and expenditures of this fund were $4.0 million and $4.9 million, respectively. The property tax rate is 23.9 mills of assessed valuation. Budgeted Fiscal Year 1990 revenues were approximately $4.5 million (HNUS 1992a).

4.4 Cultural Resources

4.4.1 Archeological Sites and Historic Structures

Field studies conducted under an ongoing program over the past two decades by the South Carolina Institute of Archeology of the University of South Carolina, under contract to DOE and in consultation with the South Carolina State Historic Preservation Officer, have provided considerable information about the distribution and content of archeological and historic resources on the SRS. By the end of Fiscal Year 1992, approximately 60 percent of the Site had been examined, and 858 archeological (historic and prehistoric) sites had been identified; these include 706 prehistoric and 350 historic components, some of which are mixed (i.e., contain elements of both). Of the 858 sites, 53 have been determined to be eligible for the National Register of Historic Places; 650 have not been evaluated. Approximately 21 of the 53 (40 percent) are historic sites, such as building foundations; none are standing structures. These sites provide knowledge of the area's history before 1820. The remainder are primarily prehistoric sites and some are mixed (historic and prehistoric). No SRS facilities have been nominated for eligibility to the National Register for Historic Places and there are no plans for such a nomination at this time (Brooks 1993; Brooks 1994). The existing SRS nuclear production facilities are not likely to be eligible for the National Register, either because they might lack architectural integrity, might not represent a particular architectural style, or might not contribute to the broad historic theme of the Manhattan Project and initial nuclear materials production (DOE 1993a).

Archeologists have divided areas of the SRS into three sensitivity zones related to their potential for containing sites with multiple archeological components or dense or diverse artifacts, and their potential for eligibility to the National Register of Historic Places (SRARP 1989).

- Zone 1 is the zone of the highest archeological site density with a high probability of encountering large archeological sites with dense and diverse artifacts, and high potential for nomination to the National Register of Historic Places.

- Zone 2 covers areas of moderate archeological site density that should contain sites of similar composition. Activities in this zone have a moderate probability of encountering archeological sites, but a low probability of encountering large sites with more than three prehistoric components. All areas within the zone are conducive to site preservation. The zone has moderate potential for encountering sites that would be eligible for nomination to the National Register of Historic Places.

- Zone 3 covers areas of low archeological site density. Activities in this zone have a low probability of encountering archeological sites and virtually no chance of encountering large sites with more than three prehistoric components; potential for site preservation is low. Some exceptions to this definition have been discovered in Zone 3, so some sites in the zone could be considered eligible for nomination to the National Register of Historic Places.

4.4.2 Native American Cultural Resources

In conjunction with 1991 studies related to a proposed New Production Reactor, DOE conducted an investigation of Native American concerns over religious rights in the Central Savannah River Valley. During this study three Native American groups - the Yuchi Tribal Organization, the National Council of Muskogee Creek, and the Indian People's Muskogee Tribal Town Confederacy - expressed concerns over sites and items of religious significance on the SRS. DOE has included these organizations on its environmental mailing list and sends them documents about SRS environmental activities (NUS 1991a).

Native American resources in the region include villages or townsites, ceremonial lodges, burial sites, cemeteries, and areas containing traditional plants for certain rituals. Villages or townsites might contain a variety of sensitive features associated with different ceremonies and rituals. The Yuchi and

Muskogee Creek tribes have expressed concerns that the area might contain several plants traditionally used in tribal ceremonies (DOE 1993a).

4.4.3 Paleontological Resources

Invertebrate fossil remains occur within the McBean, Barnwell, and Congaree formations of the Eocene Age (54 million to 39 million years ago) on the SRS. Relatively large quantities of marine invertebrate fossils have been recorded for the McBean and Barnwell Formations. Relative assessment of fossil localities is difficult because the South Carolina Geological Survey has not established criteria for, or registry of, important paleontological locations (DOE 1991b).

4.5 Aesthetics and Scenic Resources

The dominant aesthetic setting in the vicinity of the SRS consists mainly of agricultural land and forest, with some limited residential and industrial areas. Because of the distance to the Site boundary, the rolling terrain, normally hazy atmospheric conditions, and heavy vegetation, SRS facilities are not generally visible from off the Site. The few locations that have views of some of the SRS structures are quite distant from the facility [5 miles (8 kilometers) or more].

SRS land is heavily wooded, and developed areas occupy only approximately 5 percent of the total land area. The facilities are scattered across the SRS and are brightly lit at night. Typically, the reactors and principal processing facilities are large concrete structures as much as 100 feet (30 meters) high and usually colocated with lower administrative and support buildings and parking lots. The facilities are visible in the direct line-of-sight when approaching them from SRS access roads. A 500-foot cooling tower is located in K-Area. Otherwise, heavily wooded areas that border the SRS road system and public highways that cross the Site limit views of the facilities.

4.6 Geology

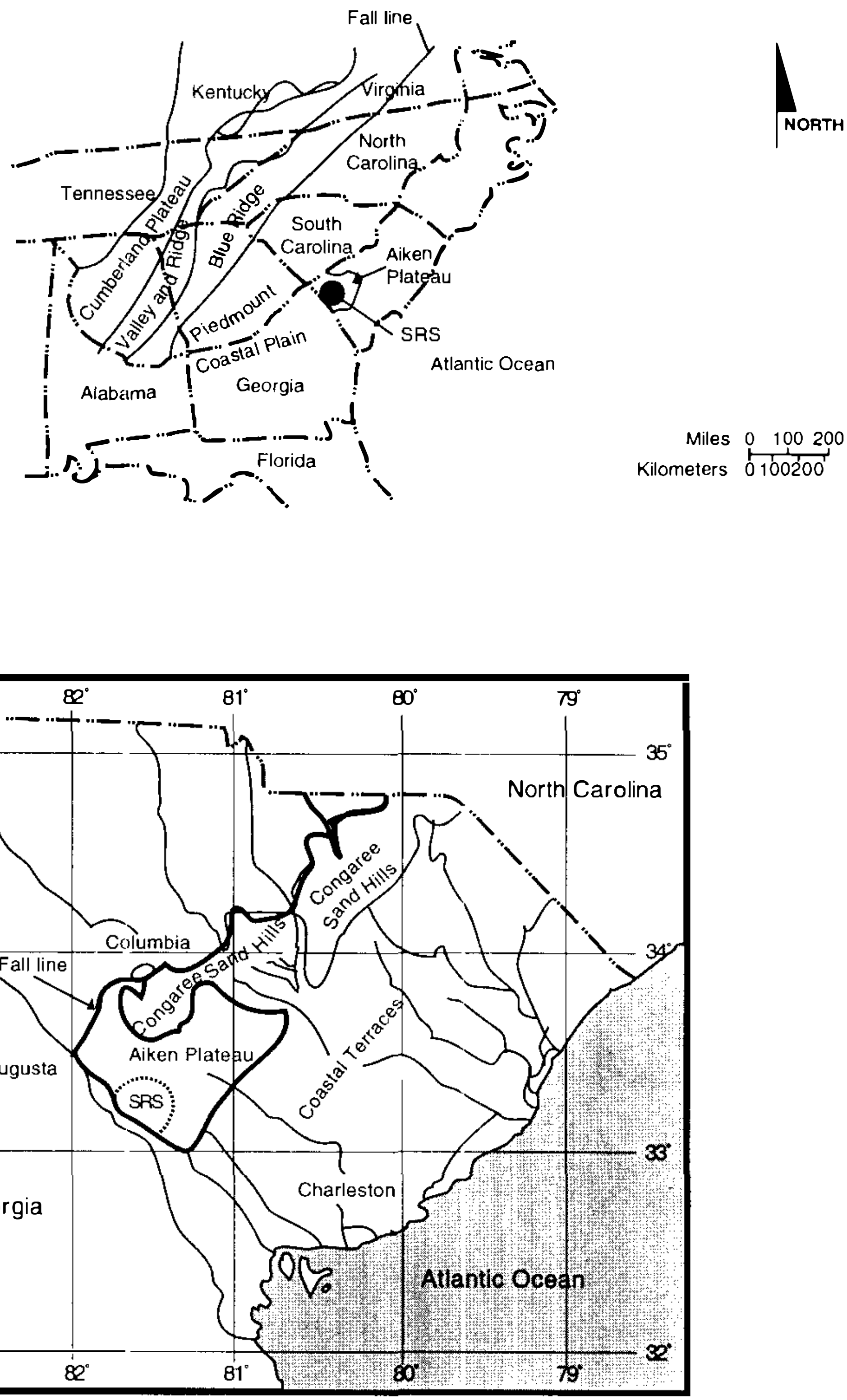
The SRS is on the Upper Atlantic Coastal Plain of South Carolina, which consists of 213 to 366 meters (700 to 1,200 feet) of sands, clays, and limestones of Tertiary and Cretaceous age. These sediments are underlain by sandstones of Triassic age and older metamorphic and igneous rocks (Arnett et al. 1993). There are no known capable faults on the SRS or volcanic activities within 800 kilometers (500 miles) of the Site.

4.6.1 General Geology

The SRS is in the Coastal Plain physiographic province of western South Carolina, approximately 32 kilometers (20 miles) southeast of the Fall Line, which separates the Piedmont and Coastal Plain provinces (Figure 4-3). The Coastal Plain province is underlain by a wedge of seaward-dipping and thickening unconsolidated and semiconsolidated sediments that extend from the Fall Line to the Continental Shelf (Figure 4-4).

In South Carolina, the Coastal Plain province is divided into the Upper Coastal Plain and the Lower Coastal Plain. Subdivisions of the Coastal Plain in the State include the Aiken Plateau and the Congaree Sand Hills in the Upper Coastal Plain, and the Coastal Terraces in the Lower Coastal Plain. The Congaree Sand Hills trend along the Fall Line northeast and north of the Aiken Plateau. The Savannah and Congaree Rivers bound the Aiken Plateau, on which the SRS is located; the plateau extends from the Fall Line to the Coastal Terraces. The surface of the plateau is highly dissected and characterized by broad interfluvial areas with narrow steep-sided valleys. The plateau is generally well drained, although poorly drained depressions (Carolina bays) do exist (DOE 1991b). Because of the proximity of the SRS to the Piedmont province, it has more relief than areas that are nearer to the coast, with onsite elevations ranging from 27 to 128 meters (89 to 420 feet) above mean sea level.

The sediments of the Atlantic Coastal Plain of South Carolina overlie a basement complex composed of Paleozoic crystalline and Triassic sedimentary rocks. These sediments dip gently seaward from the Fall Line and range in age from Late Cretaceous to Recent. The sedimentary sequence thickens from essentially zero at the Fall Line to more than 1,219 meters (4,000 feet) at the coast. Regional dip is to the southeast. Coastal Plain sediments underlying the SRS consist of sandy clays and clayey sands, although occasional beds of clean sand, gravel, clay, or carbonate occur (Figure 4-5). Two clastic limestone zones occur within the Tertiary age sequence. These calcareous zones vary in thickness from about 0.6 meter (2 feet) to approximately 24 meters (80 feet). Most of the clastic sediments are unconsolidated, but thin semiconsolidated beds also occur (DOE 1991b). Underlying sediments are dense crystalline igneous and metamorphic rock or younger consolidated sediments of the Triassic Period. The Triassic formations and older igneous and metamorphic rocks are separated hydrologically from the overlying Coastal Plain sediments by a regional aquitard, the Appleton Confining System (Arnett et al. 1993). Section 4.8.2 contains a detailed discussion of hydrogeology on the SRS.



Source: DOE (1991a)

PK54-2

Figure 4-3. Location of the Savannah River Site in the southern United States.

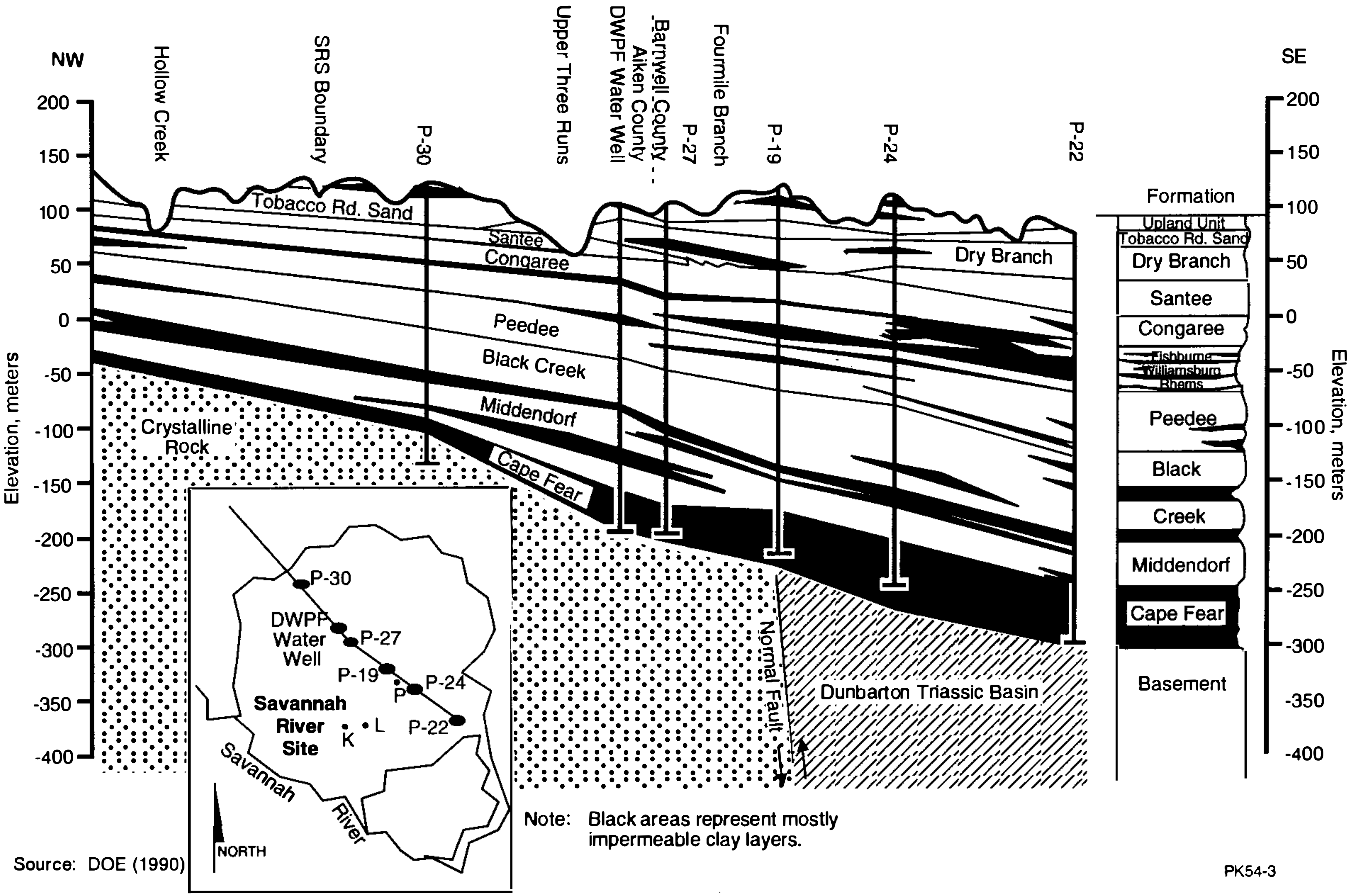


Figure 4-4. Generalized subsurface cross-section across the Savannah River Site.

Age		Unit		Lithology
Tertiary	Miocene?		"upland unit"	Clayey, silty sands, conglomerates, pebbly sands, and clays; clay clasts common
Tertiary	Eocene	Barnwell Group	Tobacco Road Sand	Red, purple, and orange, poorly to well-sorted sand and clayey sand with abundant clay laminae
Tertiary	Eocene	Barnwell Group	Dry Branch Fm.	Tan, yellow, and orange, poorly to well-sorted sand with tan and gray clay layers near base; calcareous sands and clays and limestone in lower part downdip
Tertiary	Eocene	Barnwell Group	Clinchfield Fm.	Biomoldic limestone, calcareous sand and clay, and tan and yellow sand
Tertiary	Eocene	Orangeburg Group	Santee Ls. Warley-Hill Fm.	Micritic, calcarenitic, shelly limestone, and calcareous sands; interbedded yellow and tan sands and clays; green clay and glauconitic sand near base
Tertiary	Eocene	Orangeburg Group	Congaree Fm.	Yellow, orange, tan, and greenish-gray, fine to coarse, well-sorted sand; thin clay laminae common
Tertiary	Eocene / Paleocene	Black Mingo Group	Fishburne Fm. Williamsburg Fm.	Light gray, silty sand interbedded with gray clay
Tertiary	Paleocene	Black Mingo Group	Ellenton Formation	Black and gray, lignitic, pyritic sand and interbedded clays with silt and sand laminae
Upper Cretaceous		Lumbee Group	Peedee Formation	Gray and tan, slightly to moderately clayey sand; gray red, purple, and orange clays common in upper part
Upper Cretaceous		Lumbee Group	Black Creek Formation	Tan and light to dark gray sand; dark clays common in middle and oxidized clays at top
Upper Cretaceous		Lumbee Group	Middendorf Formation	Tan and gray, slightly to moderately clayey sand; gray red, and purple clays near top
Upper Cretaceous		Lumbee Group	Cape Fear Formation	Gray, clayey sand with some conglomerates, and sandy clay; moderately to well indurated

Paleozoic Crystalline Basement
or Triassic Newark Supergroup

PK54-4

Figure 4-5. Stratigraphy of the SRS region.

4.6.2 Geologic Resources

SRS construction activities have used clay, sand, and gravel to a limited extent. These materials are not of major economic value due to their abundance throughout the region. The SRS historically has been a major user of groundwater in the region, withdrawing about 33 million liters (9 million gallons) per day. Section 4.8.2 describes the groundwater resources at the SRS.

4.6.3 Seismic and Volcanic Hazards

The closest offsite fault system of significance is the Augusta Fault Zone, approximately 40 kilometers (25 miles) from the SRS. In this fault zone, the Belair Fault has experienced the most recent movement, but it is not considered capable of generating major earthquakes (DOE 1987a). There is no conclusive evidence of recent displacement along any fault within 320 kilometers (200 miles) of the SRS, with the possible exception of the buried faults in the epicentral area of the 1886 earthquake at Charleston, South Carolina, approximately 145 kilometers (90 miles) away (DOE 1991b). Faulting in the subsurface Coastal Plain sediments in the Charleston vicinity has been suggested, based on structure contour mapping of the Eocene-Oligocene unconformity, which lies at a depth of about 30 to 61 meters (100 to 200 feet) below ground surface (WSRC 1994a). However, because it is not known if these faults offset sediments younger than Eocene-Oligocene, these shallow faults cannot be related to modem earthquakes that occur at depths greater than about 1.9 kilometers (1.2 miles). Figure 4-6 shows the geologic structures within 150 kilometers (95 miles) from the SRS, some of which are discussed above.

Several Triassic-Jurassic basins, 140 to 230 million years old, have been identified in the Coastal Plain province of South Carolina and Georgia. The Dunbarton Triassic basin, which underlies a portion of the SRS, was formed by fault movement resulting from extensional forces operating during the formation of the Atlantic Ocean. After the erosion of basin margins and infilling of the basin with Triassic age sediments, possible movement of an opposite sense to that during basin formation occurred along the fault during the Late Cretaceous age. Geophysical data indicate minimal movement on faults at the basement-Coastal Plain interface, with the exception of possible reverse fault motion along the Pen Branch Fault up into the Tertiary (WSRC 1994a).

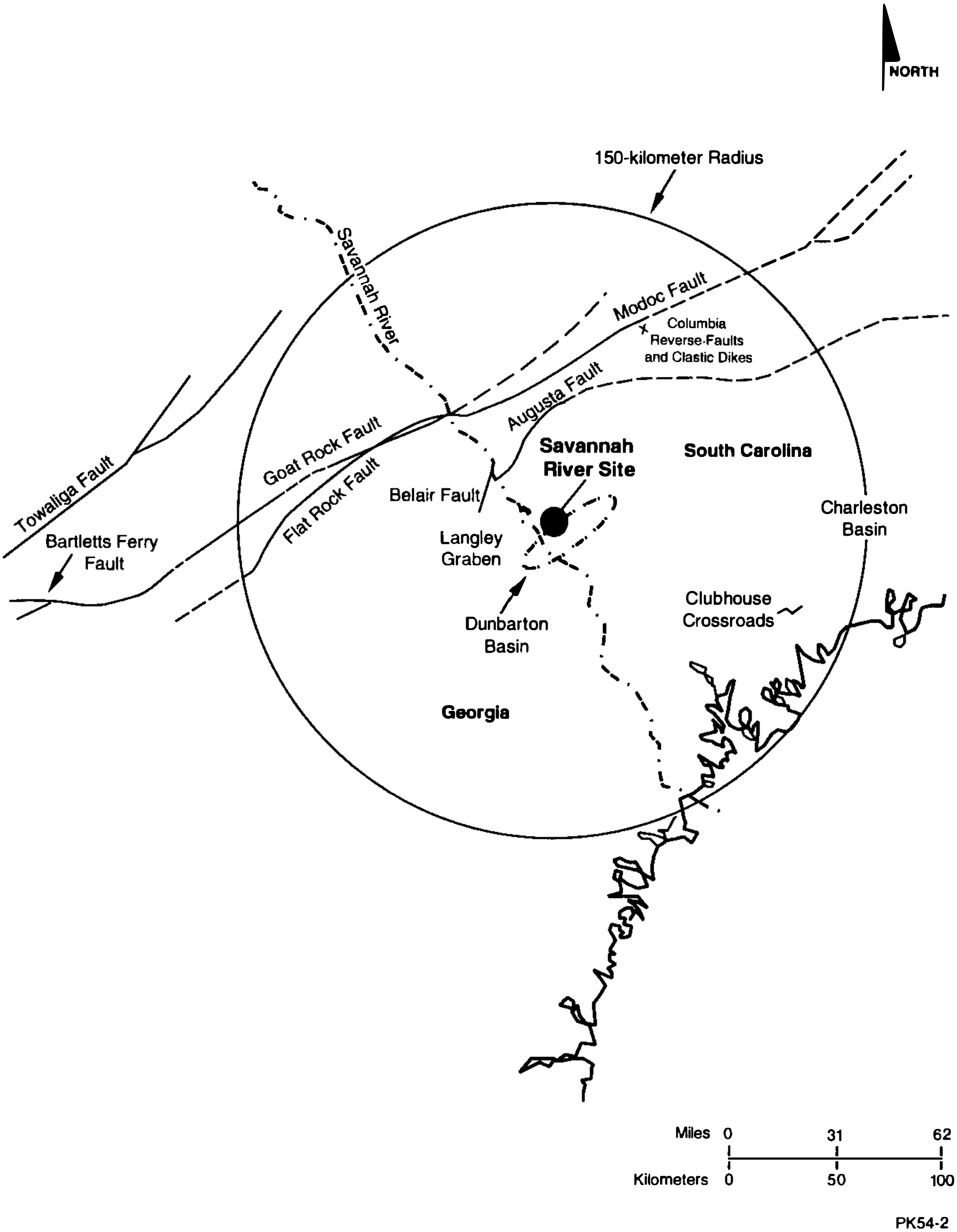


Figure 4-6. Geologic structures within 150 km of Savannah River Site.

Researchers have mapped the Pen Branch Fault for at least 24 kilometers (15 miles) across the central portion of the SRS (Snipes et al. 1993). This fault is probably a continuation of the northern boundary fault of the Triassic age Dunbarton basin and is interpreted as being at least a Cretaceous/Tertiary (144-1.6 million years) reactivation of that fault (WSRC 1994a). Observed displacements of the Coastal Plain sediments range from about 26 meters (85 feet) at the Basement/Cretaceous contact to about 9 meters (30 feet) in the shallower sediments (WSRC 1994a). Based on the available data, there is no evidence to indicate that the Pen Branch is a "capable fault" as defined by the U.S. Nuclear Regulatory Commission (NRC). Under the NRC definition, a fault is capable if it has moved within the last 35,000 years, has had recurring movement within the last 500,000 years, is related to any earthquake activity, or is associated with another capable fault. A recent study (Snipes et al. 1993) examined a Quaternary light tan soil horizon in SRS railroad cuts. The soil horizon, which has a thickness of 3 to 6 meters (10 to 20 feet), revealed no detectable offset, indicating that there has been no recent Pen Branch Fault activity. Figure 4-7 shows the locations of the Pen Branch Fault and other known or suspected faults within the Paleozoic and Triassic Basement (DOE 1991b).

Seismicity in the Coastal Plain of South Carolina occurs in three distinct seismic zones near the Charleston area (WSRC 1994a): Middleton Place-Summerville, about 19 kilometers (12 miles) northwest of Charleston; Bowman, about 59 kilometers (37 miles) northwest of the Middleton Place-Summerville; and Adams Run, about 30 kilometers (19 miles) southwest of the Middleton Place-Summerville (WSRC 1994a). Of the distinct seismic zones within the Coastal Plain province, the Charleston area has been and remains the most seismically active. The Charleston area is also the most significant source of seismicity affecting the SRS, both in terms of maximum historic site intensity and the number of earthquakes felt in the area (WSRC 1994a).

Tables 4-2 and 4-3 summarize the historic information on earthquakes that have occurred in the SRS region. Two notable earthquakes have occurred within 320 kilometers (200 miles) of the SRS. The first was a major earthquake in 1886 centered in the Charleston area about 145 kilometers (90 miles) from the Site; it had an estimated Richter magnitude of 6.8. DOE estimates that the SRS would have felt a tremor with an estimated Modified Mercalli Intensity (MMI) of VI to VII and an estimated peak horizontal acceleration of 10 percent of gravity, or $0.10g$, due to that earthquake (WSRC 1994a). The second earthquake was the Union County, South Carolina, earthquake of 1913, which had an estimated Richter magnitude of 6.0 and occurred about 160 kilometers (100 miles) from the SRS (WSRC 1994a). This earthquake, which is the closest significant event to the SRS other than

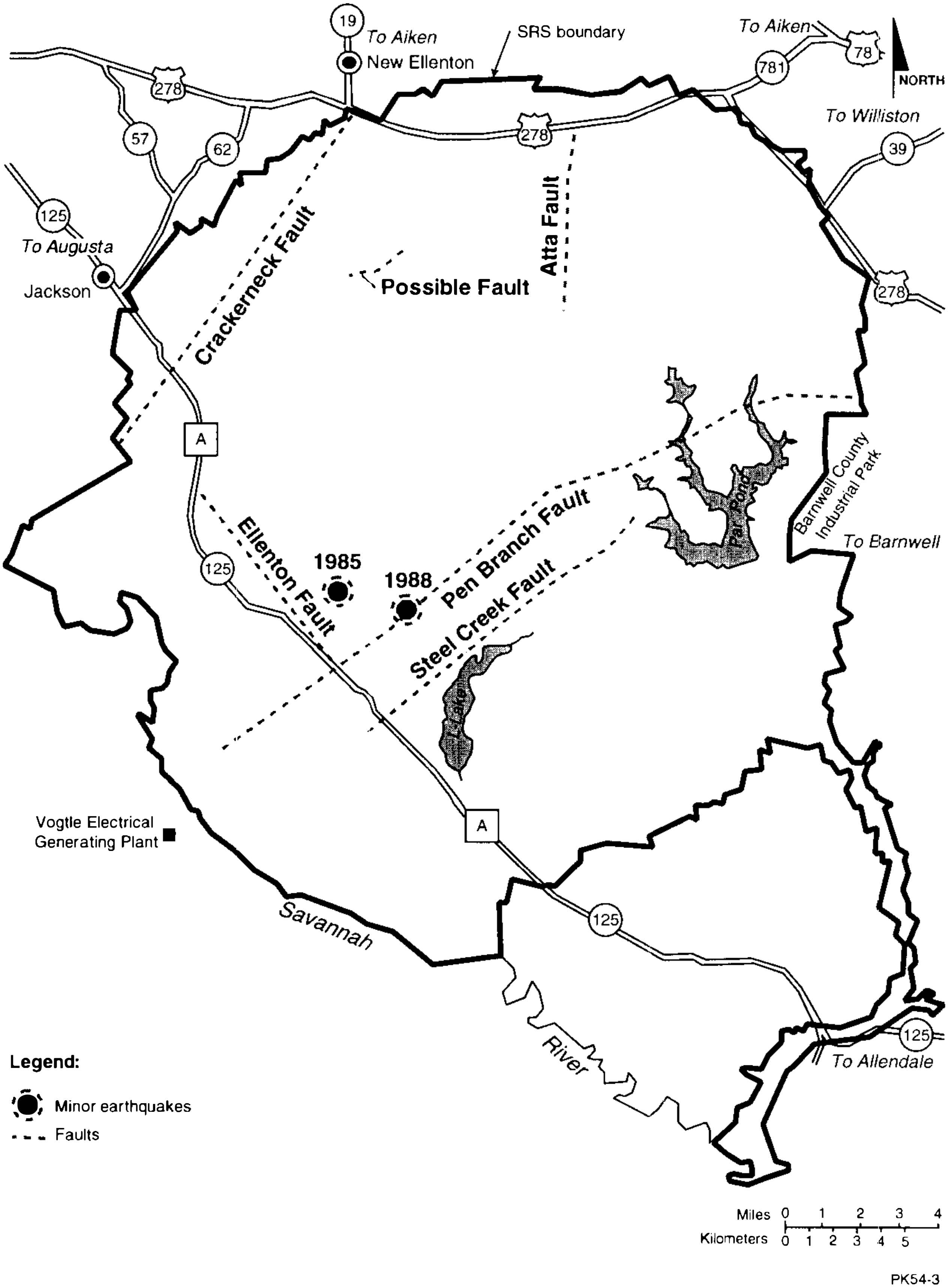


Figure 4-7. Geologic faults of the Savannah River Site.

Table 4-2. Earthquakes in the SRS region with a Modified Mercalli Intensity greater than V.[a]

Date[b]	Location	Coordinates Lat. (°N)	Coordinates Long. (°W)	Maximum Intensity	Distance from SRS (km)[c]	Reported or Estimated Intensity at SRS	Richter Magnitude	Estimated Acceleration at SRS(g)
1811 Jan 13	Burke Co., Ga.	33.2	82.2	V	55	III-IV	NA[d]	0.02
1811-1812 (3 shocks)	New Madrid, Mo.	36.3	89.5	XI-XII	850	V-VI	NA	0.05
1875 Nov 02	Lincolnton, Ga.	33.8	82.5	VI	100	III-IV	NA	0.02
1886 Sep 02	Charleston, S.C.	32.9	80.0	X	145	VI	6.8	0.10
1886 Oct 22	Charleston, S.C.	32.9	80.0	VII	155	III-IV	NA	0.02
1897 May 31	Giles Co., Va.	33.0	80.7	VIII	455	III	NA	0.02
1913 Jan 01	Union Co., S.C.	34.7	81.7	VII-VIII	160	IV	6.0[e]	0.02
1920 Aug 01	Charleston, S.C.	33.1	80.2	VII	135	III-IV	NA	0.02
1972 Feb 03	Bowman, S.C.	33.5	80.4	V	115	IV	4.5	0.02
1974 Aug 02	Willington, S.C.	33.9	82.5	VI	105	IV	4.1	0.02
1974 Nov 22	Charleston, S.C.	32.9	80.1	VI	145	III-IV	4.3	0.02

a. Source: DOE (1991b).
b. Based on Greenwich mean time.
c. Conversion factor: 1 kilometer = 0.6214 mile.
d. NA = data not available.
e. Estimated.

Table 4-3. Earthquakes in the SRS region with a Modified Mercalli Intensity greater than IV or a magnitude greater than 2.0.[a]

Date[b]	Coordinates Lat. (°N)	Coordinates Long. (°W)	Maximum Intensity	Distance from SRS (km)[c]	Reported or Estimated Intensity at SRS	Richter Magnitude	Estimated Acceleration at SRS(g)
1811 Jan 13[d]	33.2	82.2	V	55	III-IV	NA[e]	0.02
1853 May 20	34.0	81.2	VI	102	NA	NA	NA
1945 Jul 26	33.8	81.4	V	77	NA	4.4	NA
1964 Mar 07	33.7	82.4	NA	85	NA	3.3	NA
1964 Apr 20	33.8	81.1	V	96	NA	3.5	NA
1968 Sep 22	34.1	81.5	IV	102	NA	3.5	NA
1972 Aug 14	33.2	81.4	NA	27	NA	3.0	NA
1974 Oct 28	33.8	81.9	IV	72	NA	3.0	NA
1974 Nov 05	33.7	82.2	III	77	NA	3.7	NA
1976 Sep 15	33.1	81.4	NA	25	NA	2.5	NA
1977 Jun 05	3.1	81.4	NA	35	NA	2.7	NA
1982 Jan 28	32.9	81.4	NA	40	NA	3.4	NA
1985 Jun 08	33.2	81.7	III	Onsite	III	2.6	NA
1988 Feb 17[f]	33.6	81.7	III	45	NA	2.6	NA
1988 Aug 05	33.1	81.4	NA	Onsite	II	2.0	NA
1993 Aug 08	NA	NA	NA	NA	NA	3.2	NA

a. Source: DOE (1991b).
b. Based on Greenwich mean time.
c. Conversion factor: 1 kilometer = 0.6214 mile.
d. Located in Burke County, Ga.
e. NA = data not available.
f. Located at Aiken, S.C.

the Charleston-area earthquake, produced an estimated intensity of II to III (MMI) in the City of Aiken, which is approximately 19 kilometers (12 miles) north of the Site (DOE 1991b; WSRC 1994a).

Two earthquakes have occurred on the SRS during recent years (see Figure 4-7). On June 8, 1985, onsite instruments recorded an earthquake with a Richter magnitude of 2.6 and a focal depth of about 1.0 kilometer (0.6 mile) (WSRC 1994a). The epicenter was just west of the C- and K-Areas. The ground acceleration from this event did not activate instrumentation in the reactor areas (detection limits of 0.002*g*). On August 5, 1988, an earthquake with a Richter magnitude of 2.0 and a focal depth of approximately 2.7 kilometers (1.7 miles) occurred (Stephenson 1988); earthquakes of Richter magnitude 2.0 are normally detected only by specialized instrumentation. The epicenter for this event was just northeast of K-Area. Although this event was not felt by workers on the SRS, it was recorded by sensors within 96 kilometers (60 miles) of the Site. A report on the August 1988 earthquake (Stephenson 1988) also reviewed the latest earthquake history for the region. This report predicts recurrence period of 1 year for a magnitude 2.0 event for the southeast Coastal Plain. However, the report notes that historic data to calculate recurrence rates accurately are sparse. SRS workers did feel the effects of two other events that occurred in the area within the past 7 years. A Richter magnitude 2.6 earthquake occurred in the City of Aiken, approximately 19 kilometers (12 miles) north of the SRS on February 17, 1988. Reports indicate that this event was felt in the Aiken area and on the SRS (DOE 1991b). Most recently, a Richter magnitude 3.2 earthquake occurred on August 8, 1993, approximately 16 kilometers (10 miles) east of the City of Aiken near Couchton, South Carolina. Residents reported feeling this earthquake in Aiken, New Ellenton (immediately north of the SRS), North Augusta (approximately 40 kilometers [25 miles] northwest of the SRS), and the Site.

Based on seismic activity information in the past 300 years, this analysis does not project earthquakes greater than a Richter magnitude 6.0, which corresponds to a Modified Mercalli Intensity of VII, to occur on the SRS. The design-basis earthquake for the SRS is a Modified Mercalli Intensity VIII event, which corresponds to a horizontal peak ground acceleration of 0.2*g*. Based on current technology, as applied in various probabilistic evaluations of the seismic hazard in the SRS region, the 0.2*g* peak ground acceleration can be associated with a 2×10^{-4} annual probability of exceedance (5,000-year return period). DOE Standards 1020 (DOE 1994a) and 1024 (DOE 1992) summarize the results of recent seismic analyses at DOE sites and show that maximum horizontal ground accelerations for the Savannah River Site for 500 year, 1,000 year, 2,000 year, and 5,000 year seismic events are 0.10*g*, 0.13*g*, 0.18*g*, and 0.19*g* respectively. The seismic hazard information presented in this EIS is for general seismic hazard comparisons across DOE sites. Potential seismic hazards for existing and new facilities should be evaluated on a facility-specific basis consistent with DOE Orders and standards and site-specific standards.

Historically, DOE has generally selected the more conservative 0.20*g* as the peak ground acceleration for the 5,000 year seismic event when preparing safety analysis reports and environmental impact statements for the SRS. For consistency with these existing analyses, this environmental impact statement assumes 0.20*g* to be the peak horizontal ground acceleration that would result from the 5,000 year seismic event. Figure 4-8 shows seismic hazard curves for the SRS.

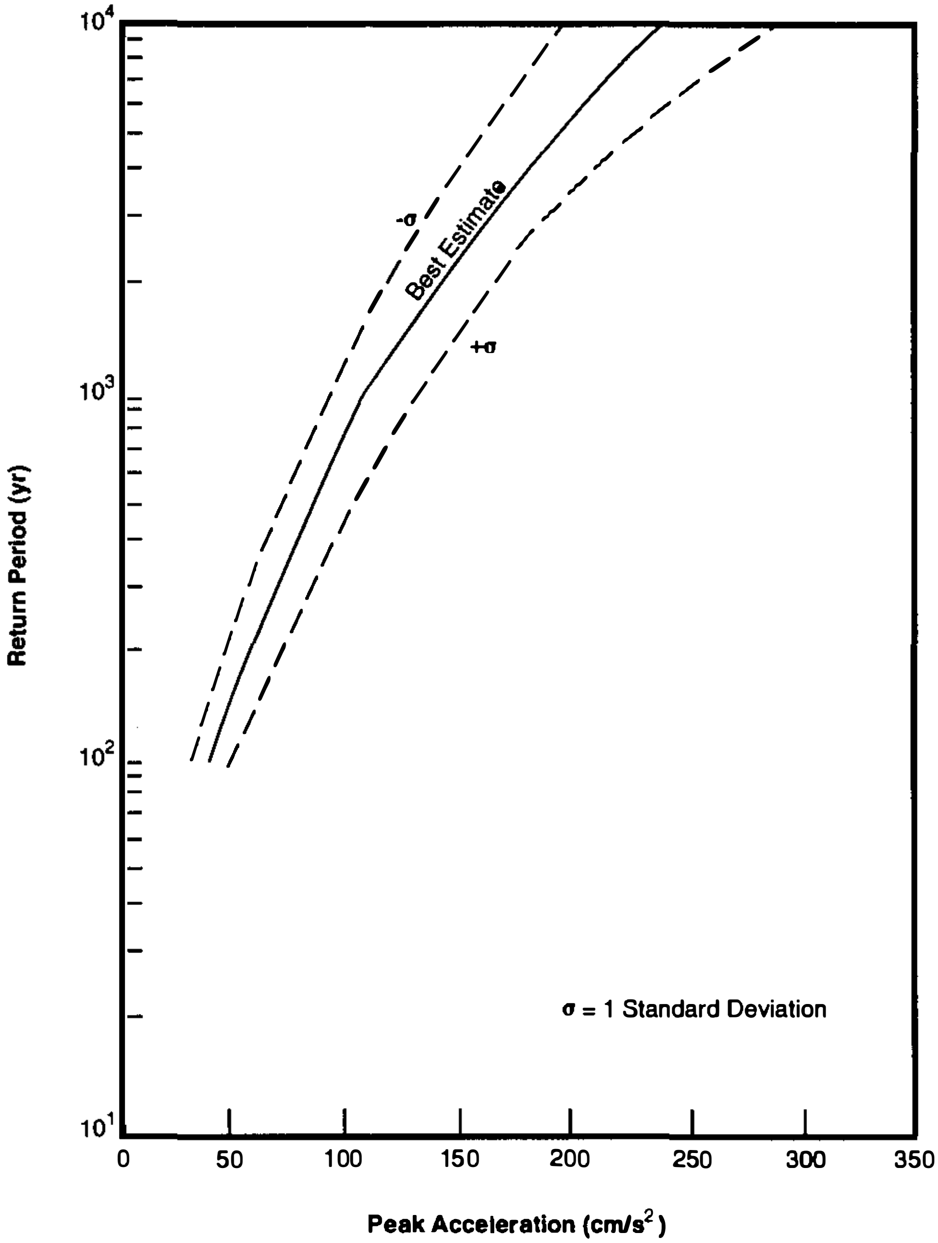
A number of paleoliquefaction sites have been identified in Beaufort County, South Carolina, some 50 miles (80 kilometers) southeast of the SRS, indicating a likelihood of prehistoric seismic events outside of the currently-active Charleston seismic zone (Rajendran and Talwani 1993). There is no evidence to suggest that seismically-induced liquefaction of soils represents a hazard at SRS, however. Weak subsurface zones are encountered occasionally during drilling. These zones are associated with carbonate materials and appear to be related to dissolution of these materials.

Engineering investigations have been conducted on granular soils underlying the Defense Waste Processing Facility [in S-Area just north of H-Area (see Figure 2-3)] to evaluate the cyclic mobility (liquefaction under cyclic stresses) of these soils (WSRC 1992b). These investigations determined that the sands and clayey sands throughout the subgrade will not experience liquefaction (strength loss leading to bearing capacity failures) and will not develop cyclic mobility (significant cyclic or accumulate deformations) under the safe shutdown earthquake with a peak horizontal ground surface acceleration of 0.20*g* (9.8 meters/second^2 or 32.1 feet/second^2).

4.7 Air Resources

4.7.1 Meteorology and Climatology

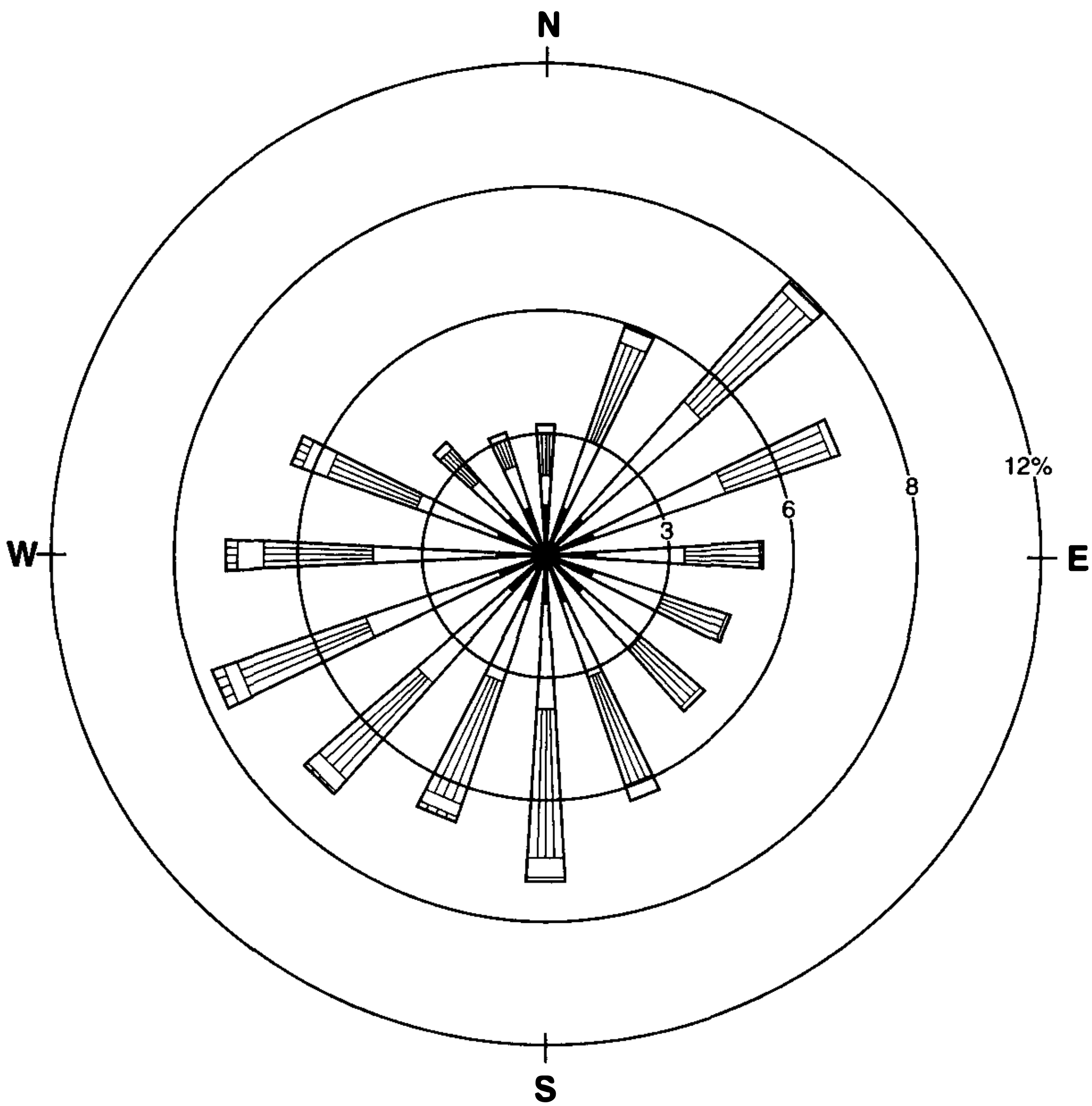
The SRS collects wind data from instruments mounted on seven onsite 61-meter (200-foot) meteorological towers. Figure 4-9 shows a wind rose that represents annual wind direction frequencies and wind speeds for the SRS from 1987 through 1991. The maximum wind directional frequencies are from the northeast and west-southwest. The average wind speed for this 5-year period was 3.8 meters per second (8.5 miles per hour). Calm winds (less than 1 meter per second or 2.2 miles per hour) occurred less than 10 percent of the time during the 5-year period. Seasonally, wind speeds



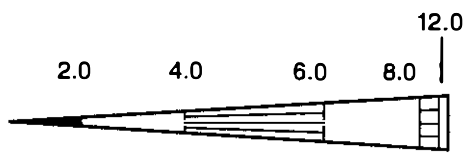
Source: Modified from Coats and Murray (1984)

PK54-2

Figure 4-8. Seismic hazard curve for SRS.



The wind rose plot shows percent occurrence frequencies of wind direction and speed at SRS. It is based on a composite of hourly averaged wind data from the SRS meteorological tower network for the 5-year period 1987-1991. Measurements were taken from 200 feet above ground. Directions indicated are *from* which the wind blows.



Wind Speed Class Boundaries
(meters/second)

PK54-4

Figure 4-9. Wind rose for the Savannah River Site (1987-1991).

were greatest during the winter at 4.1 meters per second (9.5 miles per hour) and lowest during the summer at 3.4 meters per second (7.6 miles per hour) (WSRC 1994a).

The annual average temperature at the SRS is 18 degrees C (64 degrees F); monthly averages range from a low of 7 degrees C (45 degrees F) in January to a high of 27 degrees C (81 degrees F) in July. Relative humidity readings taken four times each day range from 36 percent in April to 98 percent in August (DOE 1991a).

The average annual precipitation at the SRS is approximately 122 centimeters (48 inches). Precipitation distribution is fairly even throughout the year, with the highest precipitation in the summer [36.1 centimeters (14.2 inches)] and the lowest in autumn [22.4 centimeters (8.8 inches)]. Snowfall has occurred in the months of October through March, with the average annual snowfall at 3.0 centimeters (1.2 inches). Large snowfalls are rare (DOE 1991a).

Winter storms in the SRS area occasionally bring strong and gusty surface winds with speeds as high as 32 meters per second (72 miles per hour). Thunderstorms can generate winds with speeds as high as 18 meters per second (40 miles per hour) and even stronger gusts. The fastest 1-minute wind speed recorded at Augusta between 1950 and 1986 was 37 meters per second (83 miles per hour) (DOE 1991a).

4.7.1.1 Occurrence of Violent Weather. The SRS area experiences an average of 56 thunderstorm days per year. From 1954 to 1983, 37 tornadoes were reported for a 1-degree square of latitude and longitude that includes the SRS (DOE 1991a). This frequency of occurrence is equivalent to an average of about one tornado per year. The estimated probability of a tornado striking a point on the SRS is 7×10^{-5} per year (DOE 1991a). Since operations began at the SRS in 1953, nine confirmed tornadoes have occurred on or near the Site. They caused nothing more than light damage, with the exception of a tornado in October 1989 that caused considerable damage to forest resources in an undeveloped southeastern sector of the SRS (WSRC 1994a).

From 1700 to 1992, 36 hurricanes occurred in South Carolina, resulting in an average frequency of about one hurricane every 8 years. Three hurricanes were classified as major. Because SRS is about 160 kilometers (100 miles) inland, the winds associated with hurricanes have usually diminished below hurricane force [i.e., equal to or greater than a sustained wind speed of 33.5 meters per second (75 miles per hour)] before reaching the SRS. Winds exceeding hurricane force have been observed only once at SRS (Hurricane Gracie in 1959) (WSRC 1994a).

4.7.1.2 Atmospheric Stability. Based on measurements at onsite meteorological stations, the atmosphere in the SRS region is unstable approximately 56 percent of the time, neutral 23 percent of the time, and stable about 21 percent of the time. On an annual basis, inversion conditions occur 21 percent of the time at the SRS (WSRC 1994a).

4.7.2 Nonradiological Air Quality

4.7.2.1 Background Air Quality. The SRS is in the Augusta (Georgia) - Aiken (South Carolina) Interstate Air Quality Control Region (AQCR). This Air Quality Control Region, which is designated as a Class II area, is in compliance with National Ambient Air Quality Standards (NAAQS) for criteria pollutants. The criteria pollutants include sulfur dioxide, nitrogen oxides reported as nitrogen dioxide, particulate matter (less than or equal to 10 microns), carbon monoxide, ozone, and lead (CFR 1993a). The closest nonattainment area to the SRS is the Atlanta, Georgia, air quality region, 233 kilometers (145 miles) to the west, which is in nonattainment of the standard for ozone.

The SRS will have to comply with Prevention of Significant Deterioration (PSD) Class II requirements if there is a significant increase in emissions of criteria air pollutants due to a modification at the Site (CFR 1993b). Development at the SRS has not yet triggered Prevention of Significant Deterioration permitting requirements. If a permit were required, the SRS would have to address several requirements, including impacts on the air quality of Class I areas within 10 kilometers (6.2 miles) of the Site (CFR 1993b). The nearest Class I area to the SRS is the Congaree Swamp National Monument in South Carolina, approximately 73 kilometers (45 miles) to the east-northeast of the Site. Therefore, a Prevention of Significant Deterioration permit, if required for the SRS, would not have to address Class I areas.

4.7.2.2 Air Pollutant Source Emissions. The SRS utilized the 1990 comprehensive emissions inventory data to establish the baseline year for showing compliance with State and Federal air quality standards - calculating both maximum potential and actual emission rates. The air quality compliance demonstration also included sources forecast for construction or operation in this decade (for which the SRS had obtained air quality construction permits through December 1992). The SRS based its calculated emission rates for the sources on process knowledge, source testing, permitted operating capacity, material balance, and U.S. Environmental Protection Agency (EPA) Air Pollution Emission Factors (AP-42; EPA 1985).

4.7.2.3 Ambient Air Monitoring. At present, the SRS performs no onsite ambient air quality monitoring. State agencies operate ambient air quality monitoring sites in Barnwell, Aiken, and Richmond Counties. These areas, which include the SRS, are in attainment with National Ambient Air Quality Standards for sulfur dioxide, nitrogen oxides, carbon monoxide, particulate matter, ozone, and lead (CFR 1993a).

4.7.2.4 Atmospheric Dispersion Modeling. The SRS has performed atmospheric dispersion modeling for criteria and toxic air pollutants for both maximum potential and actual emissions for the base year 1990, using the EPA Industrial Source Complex Short Term No. 2 Model. The SRS used 1991 meteorological data collected at the Site meteorological stations for input to the model.

4.7.2.5 Summary of Nonradiological Air Quality. The SRS is in compliance with National Ambient Air Quality Standards and with the gaseous fluoride and total suspended particulate standards required by South Carolina Department of Health and Environmental Control (SCDHEC) Regulation R.61-62.5, Standard 2, "Ambient Air Quality Standards" (AAQS) (see Table 4-4).

The SCDHEC has non-radiological air quality regulatory authority over the SRS. The Department determines SRS ambient air quality compliance based on SRS air pollutant emissions modeled at the Site perimeter (excluding SC Highway 125, which crosses the southwestern quadrant of the SRS).

The SRS is in compliance with SCDHEC Regulation R.61-62.5, Standard 8, "Toxic Air Pollutants," which regulates the emission of 257 toxic substances. The SRS has identified emission sources for 139 of the 257 regulated substances; the modeled results indicate that the Site is within applicable Department of Health and Environmental Control standards (WSRC 1993a). Table 4-5 lists SRS emissions of toxic air pollutants of concern related to the SRS spent nuclear fuel alternatives, based on 1990 baseline data and the potential sources of air pollution permitted for construction or operation in December 1992.

4.7.3 Radiological Air Quality

4.7.3.1 Background and Baseline Radiological Conditions. In the SRS region, airborne radionuclides originate from natural resources (terrestrial or cosmic), worldwide fallout, and Site operations. The SRS maintains a network of air monitoring stations on and around the Site to

Table 4-4. Estimated ambient concentration contributions of criteria air pollutants from existing SRS sources and sources planned for construction or operation through 1995 ($\mu g/m^3$).[a,b]

Pollutant[c]	Averaging time	SRS Maximum Potential Concentration	Actual	Most stringent AAQS[d] (Federal or state)	Maximum Potential Concentration as a Percent of AAQS[e]
SO_2	Annual	18	10	80[f]	22.5
	24-hour	356	185	365[f,g]	97.5
	3-hour	1,210	634	1,300[f,g]	93
NO_x	Annual	30	4	100[f]	30
CO	8-hour	818	23	10,000[f,g]	8
	1-hour	3,553	180	40,000[f,g]	9
Gaseous fluorides (as HF)	12-hour	2.40	0.62	3.7[e]	65
	24-hour	1.20	0.31	2.9[e]	41
	1-week	0.6	0.15	1.6[e]	38
	1-month	0.11	0.03	0.8[e]	14
PM_{10}	Annual	9	3	50[f]	18
	24-hour	93	56	150[f]	62
O_3	1-hour	NA	NA	235[f,g]	NA
TSP	Annual geometric mean	20	11	75[e]	2.7
Lead	Calendar quarter mean	0.0015	0.0003	1.5[e]	0.1

a. Source: WSRC (1994b).
b. The contributions listed are the maximum values at the SRS boundary.
c. SO_2 = sulfur dioxide; NO_x = nitrogen oxides; CO = carbon monoxide; PM_{10} = particulate matter $\leq$ 10μm in diameter; TSP = Total Suspended Particulates, O_3 = Ozone.
d. AAQS = Ambient Air Quality Standard.
e. Source: SCDHEC (1976).
f. Source: 40 CFR Part 50.
g. Concentration not to be exceeded more than once a year.
NA = Not available.

Table 4-5. Baseline 24-hour average modeled concentrations at the SRS boundary - toxic air pollutants regulated by South Carolina from existing SRS sources and sources planned for construction or operation through 1995 ($\mu g/m^3$).[a]

Pollutant[b]	Regulatory Limit	Maximum Potential Concentration[c]	Actual Concentration[d]	Maximum Potential Concentration as a Percent of AAQS[e]
Nitric acid	125	51	4.0	41
1,1,1-Trichloroethane	9,550	81	22	1
Benzene	150	32	31	21
Ethanolamine	200	<0.01	<0.01	<0.1
Ethyl benzene	4,350	0.58	0.12	<0.1
Ethylene glycol	650	0.20	0.08	<0.1
Formaldehyde	7.5	<0.01	<0.01	<0.1
Glycol ethers	Pending	<0.01	<0.01	—
Hexachloronapthalene	1	<0.01	<0.01	<0.1
Hexane	200	0.21	0.072	<0.1
Manganese	25	0.82	0.10	3
Methyl alcohol	1,310	2.9	0.51	0.2
Methyl ethyl ketone	14,750	6.0	0.99	<0.1
Methyl isobutyl ketone	2,050	3.0	0.51	<0.1
Methylene chloride	8,750	10.5	1.8	<0.1
Naphthalene	1,250	0.01	0.01	<0.1
Phenol	190	0.03	0.03	<0.1
Phosphorus	0.5	<0.001	<0.001	<0.1
Sodium hydroxide	20	0.01	0.01	<0.1
Toluene	2,000	9.3	1.6	<0.1
Trichloroethylene	6,750	4.8	1.0	<0.1
Vinyl acetate	176	0.06	0.02	<0.1
Xylene	4,350	39	3.8	0.9

a. Source: WSRC (1994b).
b. Pollutants listed include compounds of interest regarding spent nuclear fuel alternatives.
c. Maximum potential emissions from all SRS sources for 1990 plus maximum potential emissions for sources permitted in 1991 and 1992.
d. Actual emissions from all SRS sources plus maximum potential emissions for sources permitted for construction through December 1992.
e. AAQS = Ambient Air Quality Standard.

determine concentrations of radioactive particulates and aerosols in the air (Arnett et al. 1992). Table 4-6 lists average and maximum atmospheric radionuclide concentrations at the SRS boundary and background [160-kilometer (100-mile) radius] monitoring locations during 1991. Table 4-7 lists the average concentrations of tritium in the atmosphere, as measured at on- and offsite monitoring locations.

Table 4-6. Radioactivity in air at SRS perimeter and at 160-kilometer (100-mile) radius (pCi/m^3).[a]

Location	Gross Alpha	Nonvolatile Beta	Sr-89,90[b]	Pu-238[b]	Pu-239[b]
Site perimeter					
Average	2.61×10^{-3}	1.78×10^{-2}	4.90×10^{-5}	1.22×10^{-6}	2.11×10^{-6}
Maximum	1.07×10^{-2}	4.63×10^{-2}	5.11×10^{-4}	1.94×10^{-5}	5.40×10^{-5}
Background (160-kilometer radius)					
Average	2.60×10^{-3}	1.76×10^{-2}	2.00×10^{-4}	1.44×10^{-6}	6.10×10^{-7}
Maximum	9.31×10^{-3}	5.26×10^{-2}	2.08×10^{-3}	2.39×10^{-5}	5.40×10^{-6}

a. Source: Arnett et al. (1992).
b. Monthly composite.

Table 4-7. Average atmospheric tritium concentrations on and around the Savannah River Site (pCi/m^3).[a]

Location	1991	1990	1989
Onsite	250	430	640
Site perimeter	21	32	37
40-kilometer radius	11	12	14
160-kilometer radius	8.5	8.8	9

a. Source: Arnett et al. (1992).

4.7.3.2 Sources of Radiological Emissions. Table 4-8 lists groups of facilities that released radionuclides to the atmosphere in 1992; the facilities are grouped according to the principal function that resulted in the release of radioactive materials.

Table 4-9 lists both the identified radionuclides that contributed to the SRS dose and the percent contribution of each radionuclide to the total site effective dose equivalent.

Table 4-8. Operational groupings and function of radionuclide sources.

Group	Function
Reactor Materials	Production of fuel and targets
Reactors	Irradiation of fuel and targets
Separations	Separation of useful radionuclides (other than tritium)
Analytical Laboratories	Process Control Laboratories
Tritium	Extraction, purification, and packaging
Waste Management	Management of radioactive waste
Savannah River Technology Center	Research and development to support SRS processes

4.8 Water Resources

4.8.1 Surface Water

The Savannah River bounds the SRS on its southwestern border for about 20 miles (32 kilometers), approximately 160 river miles (260 kilometers) from the Atlantic Ocean. At the SRS, river flow averages about 10,000 cubic feet (283 cubic meters) per second. River flows range from 3,960 cubic feet (112 cubic meters) per second to 71,700 cubic feet (2,030 cubic meters) per second.

Five upstream reservoirs - Jocassee, Keowee, Hartwell, Richard B. Russell, and Strom Thurmond - minimize the effects of droughts and the impacts of low flow on downstream water quality and fish and wildlife resources in the river.

At the SRS, a swamp occupies the floodplain along the Savannah River for a distance of approximately 10 miles (17 kilometers); the swamp is about 1.5 miles (2.5 kilometers) wide. A natural levee separates the river from the swampy floodplain. Figure 4-10 shows the 100-year floodplain of the Savannah River in the vicinity of the SRS as well as the floodplains of major tributaries draining the SRS. A 500-year floodplain map of the SRS has not been completed, but would be required prior to the siting of any spent nuclear fuel management facilities, in compliance with DOE regulations (CFR 1979). These regulations require DOE to evaluate the potential effects of flooding to proposed "critical actions" (for example, the storage of highly toxic or water-reactive materials), which it defines as those for which even a slight chance of flooding would be unacceptable.

The five principal tributaries to the river on the SRS are Upper Three Runs Creek, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs Creek (Figure 4-10). These tributaries drain

Table 4-9. Annual quantity of radionuclide emissions from the Savannah River Site.[a,b]

Radionuclide	Annual Quantity (curies)	Percent of Total Site Dose
H-3 (oxide)	1.00×10^{5}	98.0
Pu-239	7.45×10^{-4}	0.6
U-235,238	1.58×10^{-3}	0.4
Pu-238	4.46×10^{-4}	0.3
Ar-41	2.51×10^{2}	0.3
I-129	3.50×10^{-3}	0.2
Am-241,243	1.13×10^{-4}	0.1
Sr-89,90 (Y-90)	2.03×10^{-3}	0.02
Cm-242,244	2.31×10^{-5}	0.01
Cs-137 (Ba-137m)	2.50×10^{-4}	0.01
C-14	1.86×10^{-1}	0.01
H-3 (elemental)	5.59×10^{4}	<0.01
I-135	1.34×10^{-1}	<0.01
Kr-85	4.99×10^{1}	<0.01
I-131	9.99×10^{-5}	<0.01
Ru-106 (Rh-106)	1.81×10^{-6}	<0.01
I-133	1.15×10^{-3}	<0.01
Co-60	3.60×10^{-7}	<0.01
Xe-135	2.43×10^{-3}	<0.01
Cs-134	3.75×10^{-8}	<0.01
Ce-144 (Pr-144,144m)	1.16×10^{-7}	<0.01
Eu-154	3.44×10^{-13}	<0.01
Eu-155	1.63×10^{-13}	<0.01
Sb-125	7.27×10^{-15}	<0.01
Zr-95 (Nb-95)	2.39×10^{-14}	<0.01

a. Source: Arnett et al. (1993).
b. Includes emissions to the atmosphere and surface water.

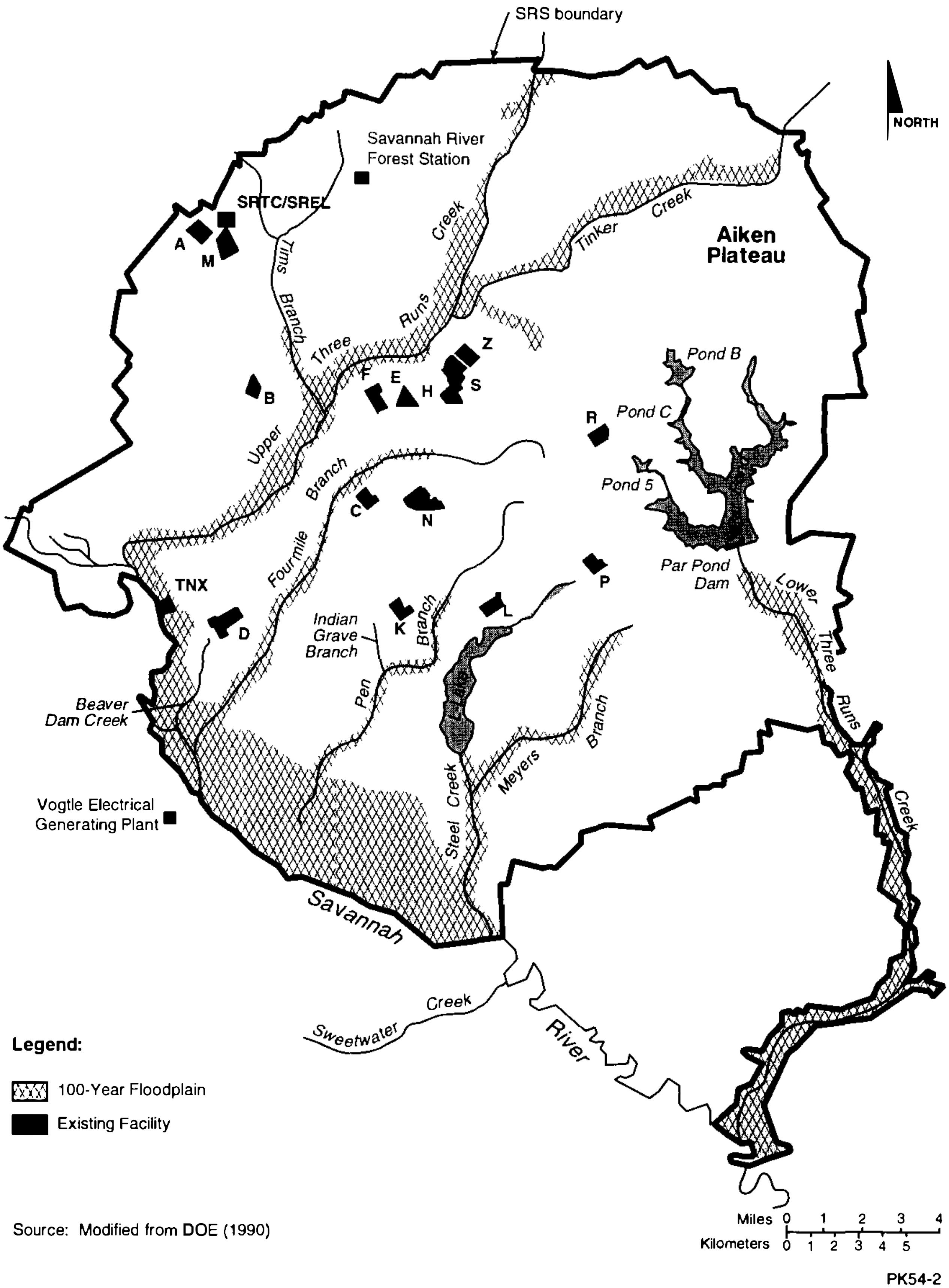


Figure 4-10. Savannah River Site, showing 100-year floodplain, major stream systems and facilities.

almost all of the SRS. Each of these streams originates on the Aiken Plateau in the Coastal Plain and descends 50 to 200 feet (15 to 60 meters) before discharging into the river. The streams, which historically have received varying amounts of effluent from various SRS operations, are not commercial sources of water. The natural flow of SRS streams ranges from less than 10 cubic feet (1 cubic meter) per second in smaller streams such as Pen Branch to 240 cubic feet (6.8 cubic meters) per second in Upper Three Runs Creek.

4.8.1.1 SRS Streams. This section describes the pertinent physical and hydrologic properties of Upper Three Runs Creek and Fourmile Branch, which are the streams closest to most SRS spent nuclear fuel management locations (Figure 4-10). These two streams are among the largest on the SRS, and they border the areas where DOE is most likely to locate new spent nuclear fuel facilities.

Upper Three Runs Creek is a large, cool [annual maximum temperature of 26.1 degrees C (79 degrees F)] blackwater stream in the northern part of the SRS. It drains an area of approximately 210 square miles (545 square kilometers), and has an average discharge of 330 cubic feet (9.3 cubic meters) per second at the mouth of the creek. Upper Three Runs Creek is approximately 25 miles (40 kilometers) long, with its lower 17 miles (28 kilometers) inside the boundaries of the SRS. This creek receives more water from underground sources than the other SRS streams and, therefore, has low conductivity, hardness, and pH values. Upper Three Runs Creek is the only major tributary on the SRS that has never received thermal discharges.

Fourmile Branch is about 15 miles (24 kilometers) long and drains an area of approximately 34 square miles (89 square kilometers). In its headwaters, Fourmile Branch is a small blackwater stream that receives relatively few impacts from SRS operations. The water chemistry in the headwater area of the creek is very similar to that of Upper Three Runs Creek, with the exception of nitrate concentrations, which are an order of magnitude higher than those in Upper Three Runs Creek (WSRC 1994a). These elevated nitrate concentrations are probably the result of groundwater transport and outcropping from the F- and H-Area seepage basins. In its lower reaches, Fourmile Branch broadens and flows through a delta formed by the deposition of sediments. Although most of the flow through the delta is in one main channel, the delta has many standing dead trees, logs, stumps, and cypress trees that provide structure and reduce the water velocity in some areas. Downstream of the delta, the creek flows in one main channel and most of the flow discharges into the Savannah River at River Mile 152 (kilometer 245), while a small portion of the creek flows west and enters Beaver Dam Creek, a small onsite tributary.

4.8.1.2 Surface Water Quality. The Savannah River, which forms the boundary between the States of Georgia and South Carolina, supplies potable water to several users. Upstream of the SRS, the river supplies domestic and industrial water needs for Augusta, Georgia, and North Augusta, South Carolina. The river also receives sewage treatment plant effluent from Augusta, Georgia; North Augusta, Aiken, and Horse Creek Valley, South Carolina; and as described above from a variety of SRS operations via onsite stream discharges. Approximately 130 river-miles (210 kilometers) downstream of the SRS, the river supplies domestic and industrial water needs for Savannah, Georgia, and Beaufort and Jasper Counties in South Carolina through intakes located at about River Mile 29 and River Mile 39. In addition, Georgia Power's Vogtle Electric Generating Plant withdraws an average of 1.3 cubic meters per second (46 cubic feet per second) for cooling and returns an average of 0.35 cubic meters per second (12 cubic feet per second) of cooling tower blowdown. Also, the Urquhart Steam Generating Station at Beech Island, South Carolina withdraws approximately 7.5 cubic meters per second (265 cubic feet per second) for once-through cooling water.

The South Carolina Department of Health and Environmental Control regulates the physical properties and concentrations of chemicals and metals in SRS effluents under the National Pollutant Discharge Elimination System (NPDES) program. This agency also regulates chemical and biological water quality standards for SRS waters. On April 24, 1992, the agency changed the classification of the Savannah River and SRS streams from "Class B waters" to "Freshwaters." The definitions of Class B waters and Freshwaters are the same, but the Freshwaters classification imposes a more stringent set of water quality standards (Arnett et al. 1993). Tables 4-10 and 4-11 list the characteristics of SRS surface-water quality upstream and downstream, respectively, due to contributions from SRS and possibly other sources. A comparison of these results indicates that influences from SRS or other sources are not seriously degrading Savannah River water quality.

4.8.2 Groundwater Resources

4.8.2.1 Hydrostratigraphic Units. There are two hydrogeologic provinces in the subsurface beneath SRS (WSRC 1994a). The first, referred to as the Piedmont hydrogeologic province (Figure 4-11), includes Paleozoic metamorphic and igneous basement rocks and Triassic-aged lithified mudstone, sandstone, and conglomerate contained within the Dunbarton Basin. The second, referred to as the Southeastern Coastal Plain hydrogeologic province, represents the major aquifer systems and consists of a wedge of unconsolidated Coastal Plain sediments of Late Cretaceous and Tertiary age (Figure 4-11). These two units are overlain by the vadose or unsaturated zone, which extends from

Table 4-10. Water quality in the Savannah River above the confluence with Upper Three Runs near the Savannah River Site in 1990.[a,b]

Parameter	Unit of Measure	MCL [c,d] or DCG[e]	Existing Water-Body Concentration[f]	
			Average	Maximum
Aluminum	mg/L	0.05-0.2[g]	NC[i]	1.1
Ammonia	mg/L	NA[j]	0.1	0.2
Cadmium	mg/L	0.005[g]	NC	<0.01
Calcium	mg/L	NA	NC	4.4
Cesium-137	pCi/L	120[e]	0.0088	0.030
Chemical oxygen demand	mg/L	NA	9.7	17
Chloride	mg/L	250[h]	7.8	11
Chromium	mg/L	0.1[d]	NC	<0.02
Copper	mg/L	1.0[d]	NC	<0.01
Dissolved oxygen	mg/L	>5	8.0	9.6
Fecal coliform	Colonies per 100/ml	1,000[g]	54	197
Gross alpha	pCi/L	15g	0.04	0.36
Iron[c]	mg/L	0.3[h]	NC	1.5
Lead	mg/L	0.015[g]	NC	0.27
Magnesium	mg/L	NA	NC	1.4
Manganese[c]	mg/L	0.05[g]	NC	0.12
Mercury	mg/L	0.002[d]	NC	<0.0002
Nickel	mg/L	0.1[c]	NC	<0.05
Nitrite/Nitrate	mg/L	10[g]	0.32	0.99
Nonvolatile beta (dissolved)	pCi/L	50[g]	1.9	3.6
pH	pH Units	6.5-8.5[g]	Not reported	7.4
Phosphate	mg/L	N/A	0.09	0.16
Plutonium-238	pCi/L	1.6[e]	0.0006	0.0021
Plutonium-239	pCi/L	1.2[e]	0.0005	0.0021
Sodium	mg/L	NA	NC	11
Strontium-89	pCi/L	800[e]	0.23	1.0
Strontium-90	pCi/L	8[c]	0.09	0.22
Sulfate	mg/L	250[h]	7.8	11
Suspended solids	mg/L	NA	13	22
Temperature	Degrees Celsius	32.2[k]	18.0	27
Total dissolved solids	mg/L	500[h]	62	76
Tritium	pCi/L	20,000[c]	150	1,110
Zinc	mg/L	5[h]	NC	0.02

a. Source: Cummins et al. (1991).
b. Parameters are those for which DOE routinely measures as a regulatory requirement or as part of ongoing monitoring programs.
c. Maximum Contaminant Level (MCL), EPA National Primary Drinking Water Regulations (CFR 1974).
d. Maximum Contaminant Level (MCL); South Carolina (1976).
e. U.S. Department of Energy Derived Concentration Guides (DCGs) for Water (DOE 1993b). DCG values are based on committed effective dose of 100 millirem per year; however, because drinking water MCL is based on 4 millirem per year, number listed is 4 percent of DCG.
f. Average concentration of samples taken at downstream monitoring station. Maximum is highest sampled concentration along reach of river potentially affected by site activities. Less than (<) indicates concentration below analysis detection limit.
g. Concentration exceeded water quality criteria; however, these criteria are listed for comparison only. Similarly, drinking water standards and DOE DCGs are listed. Water Quality Criteria (WQCs) and secondary standards are not legally enforceable.
h. Secondary Maximum Contaminant Level (SMCL), EPA National Secondary Drinking Water Regulations (CFR 1991).
i. NC = Not calculated due to insufficient number of samples.
j. NA = None applicable.
k. Shall not exceed weekly average of 32.2 degrees Celsius after mixing nor rise more than 2.8 degrees Celsius in 1 week unless appropriate temperature criterion mixing zone has been established.

Table 4-11. Water quality in the Savannah River below the confluence with Lower Three Runs near the Savannah River Site in 1990.[a,b]

Parameter	Unit of Measure	MCL [c,d] or DCG[e]	Existing Water-Body Concentration[f]	
			Average	Maximum
Aluminum	mg/L	0.05-0.2[g]	NC[i]	1.1
Ammonia	mg/L	NA[j]	0.1	0.2
Cadmium	mg/L	0.005[g]	NC	<0.01
Calcium	mg/L	NA	NC	4.4
Cesium-137	pCi/L	120[e]	0.028	0.037
Chemical oxygen demand	mg/L	NA	9.8	14
Chloride	mg/L	250[h]	8	10
Chromium	mg/L	0.1[d]	NC	<0.02
Copper	mg/L	1.0[d]	NC	<0.01
Dissolved oxygen	mg/L	>5	7.7	9.5
Fecal coliform	Colonies per 100/ml	1,000[g]	54	197
Gross alpha	pCi/L	15g	0.08	1.48
Iron[c]	mg/L	0.3[h]	NC	1.5
Lead	mg/L	0.015[g]	NC	0.01
Magnesium	mg/L	NA	NC	1.3
Manganese[c]	mg/L	0.05[h]	NC	0.1
Mercury	mg/L	0.002[d]	NC	<0.0002
Nickel	mg/L	0.1[c]	NC	<0.05
Nitrite/Nitrate	mg/L	10[g]	0.28	0.43
Nonvolatile beta (dissolved)	pCi/L	50[g]	2.1	5.1
pH	pH Units	6.5-8.5[h]	Not reported	8.2
Phosphate	mg/L	N/A	0.1	0.16
Plutonium-238	pCi/L	1.6[e]	0.0006	0.0029
Plutonium-239	pCi/L	1.2[e]	0.0014	0.0079
Sodium	mg/L	NA	NC	11
Strontium-89	pCi/L	800[e]	0.25	0.98
Strontium-90	pCi/L	8[c]	0.13	0.30
Sulfate	mg/L	250[h]	8.5	12
Suspended solids	mg/L	NA	12	19
Temperature	Degrees Celsius	32.2[k]	18.0	27
Total dissolved solids	mg/L	500[h]	63	71
Tritium	pCi/L	20,000[c]	900	6,810
Zinc	mg/L	5[h]	NC	0.02

a. Source: Cummins et al. (1991).

b. Parameters are those for which DOE routinely measures as a regulatory requirement or as part of ongoing monitoring programs.

c. Maximum Contaminant Level (MCL), EPA National Primary Drinking Water Regulations (CFR 1974).

d. Maximum Contaminant Level (MCL); South Carolina (1976).

e. U.S. Department of Energy Derived Concentration Guides (DCGs) for Water (DOE 1993b). DCG values are based on committed effective dose of 100 millirem per year; however, because drinking water MCL is based on 4 millirem per year, number listed is 4 percent of DCG.

f. Average concentration of samples taken at downstream monitoring station. Maximum is highest sampled concentration along reach of river potentially affected by site activities. Less than (<) indicates concentration below analysis detection limit.

g. Concentration exceeded water quality criteria; however, these criteria are listed for comparison only. Similarly, drinking water standards and DOE DCGs are listed. Water Quality Criteria (WQCs) and secondary standards are not legally enforceable.

h. Secondary Maximum Contaminant Level (SMCL), EPA National Secondary Drinking Water Regulations (CFR 1991).

i. NC = Not calculated due to insufficient number of samples.

j. NA = None applicable.

k. Shall not exceed weekly average of 32.2 degrees Celsius after mixing nor rise more than 2.8 degrees Celsius in 1 week unless appropriate temperature criterion mixing zone has been established.

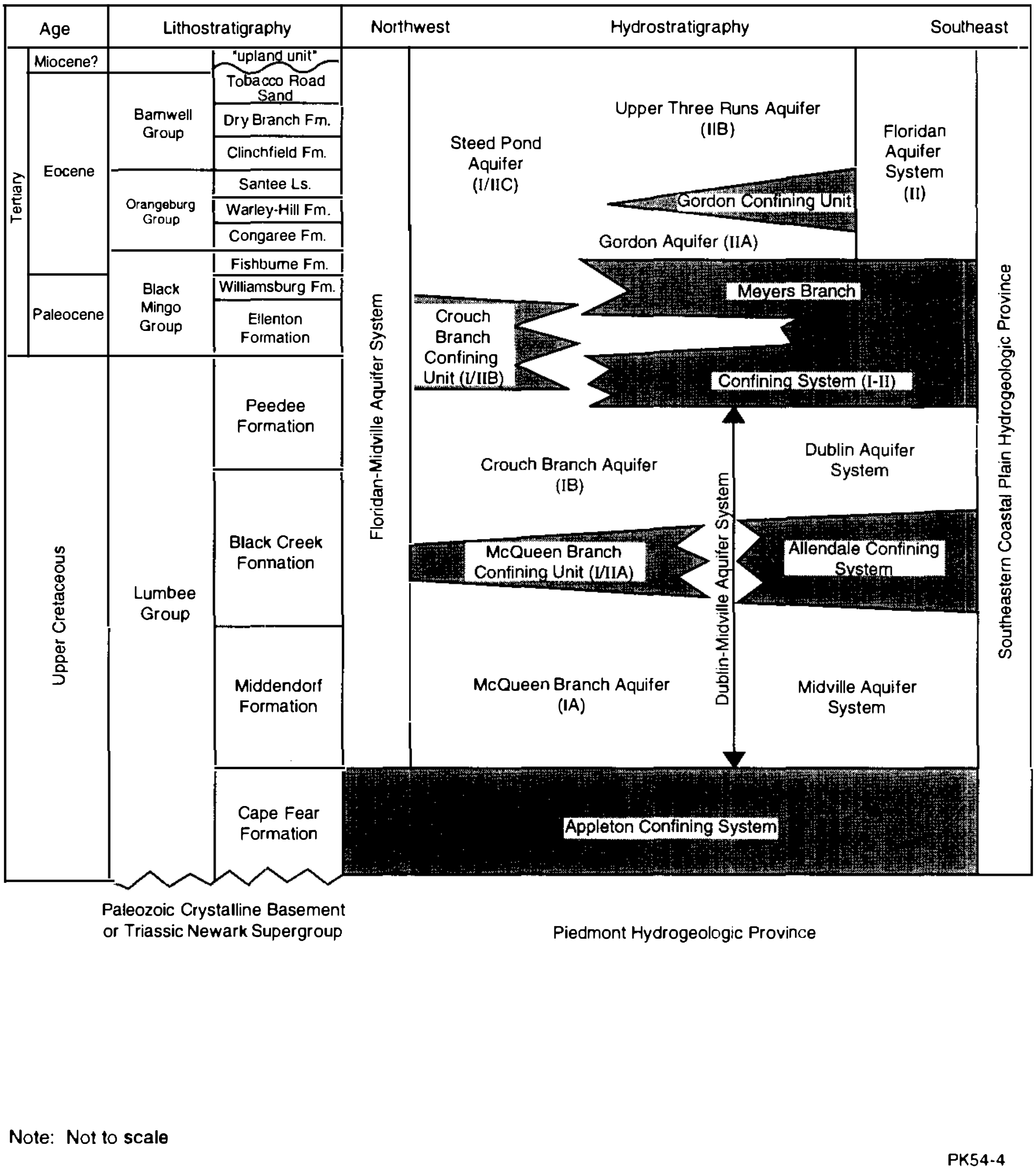


Figure 4-11. Comparison of lithostratigraphy and hydrostratigraphy for the SRS region.

the ground surface to the water table. The unsaturated zone is a heterogeneous unit of clean, clayey, or silty sand through which recharge takes place.

The sediments that make up the Southeastern Coastal Plain hydrogeologic province in west-central South Carolina are grouped into three major aquifer systems divided by two major confining systems, all of which are underlain by the Appleton confining system (Figure 4-11). The Appleton system separates the Southeastern Coastal Plain hydrogeologic province from the underlying Piedmont hydrogeologic province. Locally, each of the major aquifer systems contains individual aquifer and confining units. Figure 4-11 shows the regional lithostratigraphy of the geologic province with the attendant primary hydrostratigraphic subdivision of the province. The complexly interbedded strata that form the three aquifer systems consist primarily of fine- to coarse-grained sand and local gravel and limestone deposited under relatively high energy conditions in fluvial to shallow marine environments (WSRC 1994a).

Figure 4-11 shows the current aquifer/aquitard terminology at the SRS. Aquifers, in ascending order, include the McQueen Branch, the Crouch Branch, and the Steed Pond. For comparison, the figure also includes the corresponding aquifer terminology used on the Georgia side of the Savannah River. These include the Midville, Dublin, and Floridan aquifer systems. In addition, the three aquifers are separated by confining layers which include, in ascending order, the Appleton, Allendale, and Meyers Branch confining systems (WSRC 1994a).

4.8.2.2 Groundwater Flow. Excellent quality groundwater is abundant in this region of South Carolina from many local aquifer units. As a result, the South Carolina Department of Health and Environmental Control has classified all aquifers in the state as Class GB (South Carolina 1976), or U.S. Environmental Protection Agency (EPA) Class II, meaning that the aquifers can provide resource-quality water, but are not the sole source of supply (South Carolina Class GA or EPA Class I aquifers) (DOE 1991b).

The main source of recharge to the vadose zone is rainfall. The annual precipitation at the SRS is 48 inches (121.9 centimeters), with an estimated 16 inches (41 centimeters) designated as surface recharge at the center of the SRS, in bare and grass-covered areas (WSRC 1994a). The direction of groundwater flow in the vadose zone is predominantly downward. However, given the lenses of silt and clay that exist, there is significant lateral spread in some areas. In general, the vadose zone thickness ranges from approximately 130 feet (40 meters) in the northernmost portion of the SRS to 0 feet where the water table intersects wetlands, streams, or creeks.

The following discussion of groundwater flow in the Coastal Plain hydrogeologic province begins with the deepest aquifers at the SRS and proceeds to shallower units. It does not address flow in the confining units because few hydraulic head measurements are available for these units and, to a good approximation, flow in aquitards is limited predominantly to vertical flow between aquifer units. The Midville or McQueen Branch aquifer (which has also been called the Middendorf, the Lower Cretaceous, the Tuscaloosa, and Aquifer IA) is highly transmissive and, therefore, serves in part as the production aquifer for much of the SRS. This aquifer flows horizontally, predominantly toward the Savannah River. In the past, groundwater production wells at the SRS were screened in both the Midville (McQueen Branch) and Dublin (Crouch Branch) aquifers. In 1985 DOE committed to the South Carolina Department of Health and Environmental Control to complete production wells only in the McQueen Branch aquifer to minimize the potential for contamination to reach such wells and spread in the deeper aquifers.

Flow in the Dublin or Crouch Branch aquifer (which has also been called the Black Creek, the Tuscaloosa, the Upper Cretaceous, and Aquifer IB) is more complicated than flow in the deeper McQueen Branch aquifer because of the apparent communication with Upper Three Runs Creek on the SRS. Nonetheless, horizontal flow in the Dublin (Crouch Branch) aquifer is predominantly toward the Savannah River. However, there is an upward vertical flow component near the river and Upper Three Runs Creek. Recharge to the Dublin-Midville aquifer system occurs in areas exposed at the ground surface near the Fall Line (see Figure 4-3).

Horizontal flow in the Gordon aquifer (previously called the Congaree, the Tertiary, and Aquifer II) is toward Upper Three Runs Creek and the Savannah River, depending on the area of the SRS. Both the river and Upper Three Runs Creek intercept this aquifer. The Gordon aquifer receives most of its recharge from groundwater that originates on the SRS.

Previous SRS studies have called the Upper Three Runs aquifer the "water table aquifer"; others have defined it as both the Barnwell/McBean and water table aquifers in the central portion of the SRS where those aquifers were thought to be separated by a "tan clay." The Upper Three Runs aquifer is the shallowest aquifer at the SRS. The horizontal groundwater flow is generally toward the nearest surface-water feature that is in communication with the water table. Most SRS streams, except Tims Branch in the northeastern part of the Site, are in communication with the water table. Tims Branch is a "losing stream," meaning it provides, or "loses," water to the Upper Three Runs aquifer. However, the Upper Three Runs aquifer receives most of its recharge from precipitation. The Upper Three Runs aquifer is not a source of domestic or production water on the SRS because the lower aquifers provide a more abundant supply of higher quality water (WSRC 1994a).

4.8.2.3 Groundwater Quality. The quality of groundwater in the principal hydrologic systems beneath the SRS depends on both the source of the water and the inorganic and biochemical reactions that take place along its flowpath. Quality is strongly influenced by the chemical composition and mineralogy of the enclosing geologic materials (WSRC 1994a).

In general, the quality of the groundwater in the Coastal Plain sediments at the SRS and the surrounding areas is suitable for most domestic and industrial purposes. The waters have low concentrations of total dissolved solids (TDS), ranging from less than 10 milligrams per liter to about 150 to 200 milligrams per liter. The pH values range from 4.9 to 7.7 (where the groundwater is in contact with limestone). Much of the groundwater is corrosive to metal surfaces due to its low solids content and frequently low pH values. High dissolved iron concentrations can also be of concern in some groundwater units. The SRS uses degasification and filtration processes to raise the pH and remove iron in domestic water supplies where necessary (WSRC 1994a).

Table 4-12 summarizes groundwater quality data from 85 existing waste sites on the SRS compared to drinking water standards; Table 4-13 lists similar information for selected radiological constituents. The data in these tables are from ongoing monitoring programs on the Site. EPA-accepted methods and guidelines for sampling and analysis are an integral part of this monitoring program. Several of the facilities discussed below have state-approved sampling and analysis plans.

The shallow aquifers beneath 5 to 10 percent of the SRS have been contaminated by industrial solvents, metals, tritium, or other constituents used or generated on the Site. Figure 4-12 shows the locations of facilities where the SRS monitors groundwater and areas with constituents that exceeded drinking water standards in 1992; the concentrations shown on Figure 4-12 represent the maximum data from one monitoring well on at least one occasion at a given area. Contamination is limited to the shallow aquifers, with one exception (see next paragraph). Most contaminated groundwater at the SRS is beneath a few facilities; contaminants reflect the operations and chemical processes those facilities perform. For example, contaminants in the groundwater beneath A- and M-Areas include chlorinated volatile organics, radionuclides, metals, and nitrate. At F- and H-Areas, contaminants in the groundwater include tritium and other radionuclides, metals, nitrate, chlorinated volatile organics at values much smaller than those found at A- and M-Areas, and sulfate. The groundwater beneath the Sanitary Landfill contains chlorinated volatile organics, radionuclides, and metals. The groundwater

Table 4-12. Representative groundwater quality data for nonradioactive constituents from the Savannah River Site.[a]

Parameter (Unit)	Standard	Maximum Value
Alkalinity (as $CaCO_3$) (mg/L)	100	1,360[b]
pH (pH units)	8.5[c]	13[b]
Antimony (mg/L)	0.005	0.013
Arsenic (mg/L)	0.05	0.1
Beryllium (mg/L)	0.011[d]	0.0043
Cadmium (mg/L)	0.005[c]	0.34
Chromium (mg/L)	0.1[c]	0.82
Mercury (mg/L)	0.002[c]	0.12
Lead (mg/L)	0.015[e]	1.0
Nitrate-N (mg/L)	10[c]	278[b]
Sulfate (mg/L)	400[c]	73,500[b]
Pentachlorophenol (mg/L)	0.001[c]	0.0032
Lindane (mg/L)	0.0002[c]	0.00048
Carbon tetrachloride (mg/L)	0.005	0.43
1,2-Dichloroethane (mg/L)	0.005[c]	0.27
1,1,1-Trichloroethane (mg/L)	0.2[c]	0.21
1,1-Dichloroethylene (mg/L)	0.007[c]	0.15
Trichlorethylene (mg/L)	0.005[c]	147
Tetrachloroethylene (mg/L)	0.005[c]	101

a. Data compiled from 85 existing wastes sites (Arnett et al. 1993).
b. The elevated values for alkalinity and pH might be due to faulty well installation; the elevated sulfate and nitrate values might be due to acid spills near wells.
c. National secondary drinking water regulations (CFR 1991).
d. National primary drinking water regulations (CFR 1974).
e. Action level at which providers of public drinking water apply treatment technique to reduce lead levels (CFR 1991).

Table 4-13. Representative groundwater data for radioactive constituents from the Savannah River Site (pCi/liter).[a]

Constituent	Standard[b]	Maximum Concentration
Gross alpha	15	2,700
Nonvolatile beta	50	19,000
Tritium	20,000	1.8×10^8
Cesium-137	200	980
Cobalt-60	100	290
Iodine-129	1	72
Ruthenium-106	30	170
Total radium (radium-226 and radium-228)	5	50
Strontium-90	8	5,300

a. Source: Arnett et al. (1993).
b. National Primary Drinking Water Regulations (CFR 1974), (56 FR 33052).

beneath all the reactor areas except R-Area contains tritium, other nuclides, metals, and chlorinated volatile organics. At R-Area, groundwater contaminants include radionuclides and cadmium. The groundwater beneath D-Area contains metals, radionuclides, sulfate, and chlorinated volatile organics. At TNX-Area, the groundwater contains chlorinated volatile organics, radionuclides, and nitrate (Arnett et al. 1993). None of these cases indicated the presence of groundwater contamination beyond Site boundaries. With the ongoing and expanding "pump and treat" system at the A-/M-Area (Figure 4-12), concentrations in the volatile organic compound plume are likely to decrease with time.

Contamination of groundwater in a drinking water aquifer has been found in only one relatively-small area north of A-Area, in the northwest portion of the site. In the early 1980s, SRS monitors found low concentrations of trichloroethylene (11.7 microgram per liter) in water from one production well (53A) completed to the Dublin-Midville Aquifer System (formerly called the Tuscaloosa Formation) in M-Area. The monitors found the contamination only at 430 and 480 feet (131 and 146 meters) in this well, which is 670 feet (204 meters) deep. The well is screened intermittently from 387 feet (118 meters) to the bottom. DOE concluded that the contamination is probably migrating down the outside well casing from soils near the surface that are contaminated with trichloroethylene. This contaminated water enters the well through screens set in the Dublin-Midville

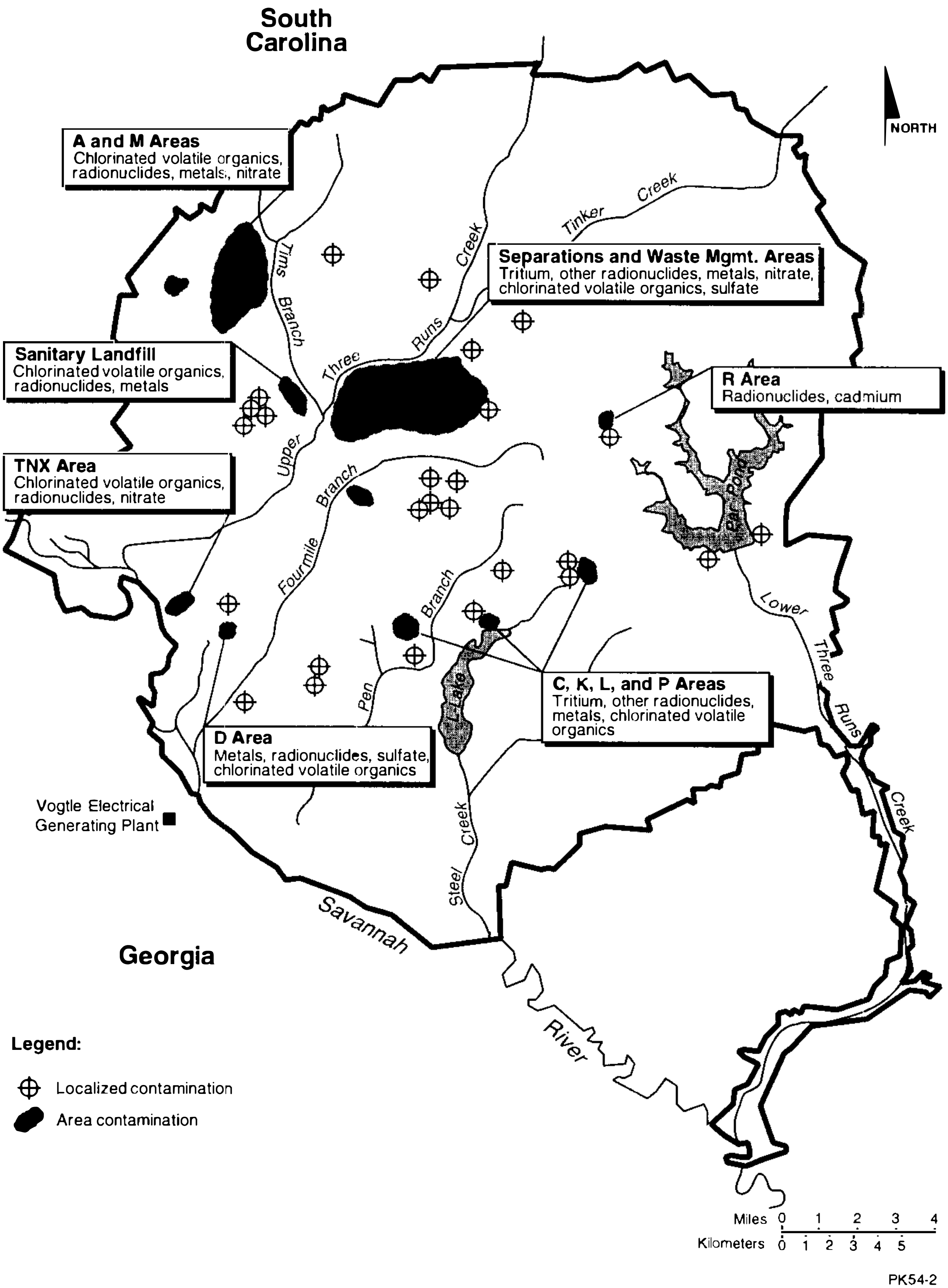


Figure 4-12. Groundwater contamination at the Savannah River Site.

System (Du Pont 1983). In addition, in 1992 trichloroethylene and tetrachloroethylene were detected above Primary Drinking Water Standards in cretaceous zone (Dublin-Midville) well MSB 55TA, which is approximately 3,500 feet west of well 53A and 1,500 feet north of A-Area (Arnett et al. 1993).

4.8.2.4 Groundwater Use. The McQueen Branch aquifer, which becomes shallower toward the Fall Line, forms the base for most municipal and industrial water supplies in Aiken County. Toward the coast, in Allendale and Barnwell Counties, this aquifer exists at increasingly greater depths. As a consequence, the shallower Gordon aquifer supplies some municipal, industrial, and agricultural users (Arnett et al. 1993).

DOE has identified 56 major municipal, industrial, and agricultural groundwater users within 20 miles (32 kilometers) of the center of the SRS (DOE 1987a). The total pumpage for these users is about 49 billion liters (13 billion gallons) per year. The SRS withdraws approximately 14.0 billion liters (3.7 billion gallons) of groundwater per year for domestic and industrial uses (DOE 1990).

4.9 Ecological Resources

The U.S. Government acquired the SRS in 1951. At that time, the Site was approximately two-thirds forested and one-third cropland and pasture (Dukes 1984). At present, more than 90 percent of the SRS is forested. An extensive forest management program conducted by the Savannah River Forest Station, which is operated by the U.S. Forest Service, has converted many pastures and croplands to pine plantations. With the exception of the SRS production and support areas, natural succession has reclaimed previously disturbed areas. Table 4-14 lists SRS land cover, other than the land used for nuclear reactors and support facilities.

The SRS is important to maintaining the biodiversity of the region. Satellite imagery of the Site shows a circle of wooded habitat within a matrix of cleared uplands and narrow forested riparian corridors. The SRS provides more than 734 square kilometers (181,000 acres) of contiguous forested cover broken only by unpaved secondary roads, transmission line corridors in various stages of succession, and a few paved primary roads. Carolina bays, the Savannah River swamp, and several relatively intact longleaf pine-wiregrass communities provide important contributions to the biodiversity of the SRS and of the entire region.

Table 4-14. Land cover of undeveloped areas on the Savannah River Site.[a]

Land cover types	Square kilometers[b]	Percent of total
Longleaf pine	150	20
Loblolly pine	258	35
Slash pine	117	16
Mixed pine/hardwood	23	3
Upland hardwood	20	3
Bottomland hardwood	117	16
Savannah River swamp	49	7
Total	734	100.0

a. Source: USDA (1991a).
b. To convert square kilometers to acres, multiply by 247.1.

F- and H-Areas, located near the center of the SRS and approximately 1.6 kilometers (1 mile) southeast of Upper Three Runs Creek, are heavily industrialized with little natural vegetation remaining inside the fenced areas. These areas are dominated by buildings, paved parking lots, gravelled construction areas, and laydown yards. While some grassed areas occur around the administration buildings and some vegetation is present along the ditches that drain the area, the majority of the site contains no vegetation. Wildlife is absent except for occasional crows (*Corvus brachyrhynchos*) and nesting barn swallows (*Hirundo rustica*) around the buildings.

Figure 2-3 shows the location of a representative host site at the SRS for potential spent nuclear fuel activities. F- and H-Areas (and developed areas immediately adjacent to them) would house most spent nuclear fuel management facilities, while the undeveloped area south and east of H-Area would be used for the construction of new facilities that F- and H-Areas could not accommodate. The undeveloped area, which was 98 percent cleared fields in 1951, is now almost completely forested, for the most part with 5- to 40-year-old upland pine stands that are actively managed by the Savannah River Forest Station. Most of these stands are loblolly pine (*Pinus taeda*), but there are small stands of slash pine (*P. elliottii*), upland hardwoods (predominantly oaks and hickories), and bottomland hardwoods (most commonly sweetgum, *Liquidambar styraciflua*, and yellow poplar, *Liriodendron tulipifera*) associated with two small Carolina bays located south of H-Area. The area south of H-Area lies in the Fourmile Branch watershed, while the area east of H-Area is in the McQueen Branch (a tributary of Upper Three Runs Creek) watershed. Neither area is likely to contain any threatened or endangered species or their habitats.

The general area of the representative host site contains suitable habitat for white-tailed deer and feral hogs as well as other faunal species common to the mixed pine/hardwood forests of South Carolina. Additional wildlife species found in the area include gray squirrel (*Sciurus carolinensis*), fox squirrel (*S. niger*), wild turkey (*Meleagris gallopovo*), cottontail rabbit (*Sylvilagus floridanus*), raccoon (*Procyon lotor*), bobcat (*Felix rufus*), and gray fox (*Urocyon cinereoargenteus*).

4.9.1 Terrestrial Ecology

The SRS is near the transition area between the oak-hickory-pine forest and the southern mixed forest. As a consequence, species typical of both associations occur (Dukes 1984). In addition, farming, fire, soil features, and topography have strongly influenced existing SRS vegetation patterns.

A variety of vascular plant communities occurs in the upland areas (Dukes 1984). Typically, scrub oak communities occur on the drier, sandier areas. Longleaf pine (*Pinus palustrus*), turkey oak (*Quercus laevis*), bluejack oak (*Q. incana*), blackjack oak (*Q. marilandica*), and dwarf post oak (*Q. margaretta*) dominate these communities, which typically have understories of wire grass (*Aristida stricta*) and huckleberry (*Vaccinium* sp.). Oak-hickory communities occur on more fertile, dry uplands; characteristic species are white oak (*Q. alba*), post oak (*Q. stellata*), southern red oak (*Q. falcata*), mockernut hickory (*Carya tomentosa*), pignut hickory (*C. glabra*), and loblolly pine, with an understory of sparkleberry (*Vaccinium arboreum*), holly (*Ilex* sp.), greenbriar (*Smilax* sp.), and poison ivy (*Rhus radicans*).

The removal of human residents in 1951 and the subsequent restoration of forest cover has provided the wildlife of the SRS with excellent habitat. Furbearers such as gray fox, raccoon, opossum (*Didelphis virginiana*), bobcat, beaver (*Castor canadensis*), and otter (*Lutra canadensis*) are relatively common throughout the Site. Game species such as gray squirrel and fox squirrel, white-tailed deer (*Odocoileus virginianus*), cottontail rabbit, and wild turkey are also common. The Savannah River Ecology Laboratory has conducted numerous studies of reptile and amphibian use of the wetlands and adjacent uplands of the SRS.

DOE allows carefully regulated public hunting for white-tailed deer and feral hogs (*Sus scrofa*) on most of the SRS to reduce the incidence of animal/vehicle collisions and maintain healthy populations within the carrying capacity of the range. SRS personnel monitor all animals removed from the Site for contamination before releasing them to the hunters (WSRC 1992a).

Before releasing any animal to a hunter, SRS technicians perform field analyses for cesium-137 at the hunt site. In 1992, hunters collected 1,519 deer and 168 hogs. The maximum 1992 cesium-137 field measurement for deer was 22.4 picocuries per gram; the average was 6.4 picocuries per gram (Arnett et al. 1993). For hogs, the maximum value was 22.9 picocuries per gram and the average was 3.5 picocuries per gram. The field technicians determine estimated doses from consumption of the venison and pork and make this information available to the hunters.

In 1992, the estimated maximum dose received by a hunter was 49 millirem per year. The basis for this unique hypothetical maximum dose, which was for a hunter who harvested eight deer and one hog, is the assumption that the hunter consumed the entire edible portion of each animal. An additional hypothetical model involved a hunter whose total meat consumption for the year consisted of SRS deer [81 kilograms (179 pounds) per year] (Arnett et al. 1993). Based on these low-probability assumptions and on the average concentration of cesium-137 (6.4 picocuries in deer harvested on the SRS), the estimated potential maximum dose from this pathway is 26 millirem; this is 26 percent of the annual 100-millirem DOE Derived Concentration Guide. Although a large percentage of this hypothetical dose is probably due to cesium-137 from worldwide fallout, the estimated total contains this background cesium-137 for conservatism.

4.9.2 Wetlands

The SRS has extensive, widely distributed wetlands, most of which are associated with floodplains, creeks, and impoundments. In addition, approximately 200 Carolina bays occur on the Site (Shields et al. 1982; Schalles et al. 1989).

The southwestern SRS boundary adjoins the Savannah River for approximately 32 kilometers (20 miles). The river floodplain supports an extensive swamp, covering about 49 square kilometers (12,148 acres) of the Site; a natural levee separates the swamp from the river. Timber was cut in the swamp in the late 1800s. At present, the swamp forest consists of second-growth bald cypress (*Taxodium distichum*), black gum (*Nyssa sylvatica*), and other hardwood species (Workman and McLeod 1990; USDA 1991a).

Five major streams drain the SRS and eventually flow into the Savannah River. Each stream has floodplains characterized by bottomland hardwood forests or scrub-shrub wetlands in varying stages of succession. Dominant species include red maple (*Acer rubrum*), box elder (*A. negundo*), bald cypress, water tupelo (*Nyssa aquatica*), sweetgum, and black willow (*Salix nigra*) (Workman and McLeod 1990).

Carolina bays are unique wetland features of the southeastern United States. They are islands of wetland habitat dispersed throughout the uplands of the SRS. The approximately 200 bays on the Site exhibit extremely variable hydrology and a range of plant communities from herbaceous marsh to forested wetland (Shields et al. 1982; Schalles et al. 1989). SRS scientists have studied Carolina bay ecology extensively, particularly in relation to the construction of the Defense Waste Processing Facility (DWPF; SREL 1980).

4.9.3 Aquatic Ecology

The aquatic resources of the SRS have been the subject of intensive study for more than 30 years. Research has focused on the flora and fauna of the Savannah River and the five tributaries of the river that drain the Site. Section 4.8.1.1 describes those portions of the aquatic systems that spent nuclear fuel management activities could affect. In addition, several monographs (Patrick et al. 1967; Dahlberg and Scott 1971; Bennett and McFarlane 1983), the eight-volume Comprehensive Cooling Water Study (Du Pont 1987), and three EISs (DOE 1984; DOE 1987b; DOE 1990) that evaluated operations of SRS production reactors describe the aquatic biota and aquatic systems of the SRS.

4.9.4 Threatened and Endangered Species

Threatened, Endangered, and Candidate Plant and Animal Species of the Savannah River Site (HNUS 1992b) describes threatened, endangered, and candidate plant and animal species that are known to occur or that might occur on the SRS. Table 4-15 lists these species.

The following Federally listed endangered animals are known to occur on the SRS or in the Savannah River adjacent to the Site: the red-cockaded woodpecker (*Picoides borealis*), the southern bald eagle (*Haliaeetus leucocephalus*), the wood stork (*Mycteria americana*), and the shortnose sturgeon (*Acipenser brevirostrum*) (HNUS 1992b). Researchers have found one Federally listed endangered plant species, the smooth coneflower (*Echinacea laevigata*), on the Site, several Federally

Table 4-15. Threatened, endangered, and candidate plant and animal species of the SRS.

Common Name (Scientific Name)	Status
Animals	
Rafinesques (= Southeastern) big-eared bat (*Plecotus rafinesquii*)	FC2
Loggerhead Shrike (*Lanius ludovicianus*)	FC2
Bachman's sparrow (*Aimophila aestivalis*)	FC2
Carolina crawfish (= Gopher) frog (*Rana areolata capito*)	FC2
Southern hognose snake (*Heterodon simus*)	FC2
Northern pine snake (*Pituophis melanoleucus melanoleucus*)	FC2
Bald eagle (*Haliaeetus leucocephalus*)	E
Wood stork (*Mycteria americana*)	E
Red-cockaded woodpecker (*Picoides borealis*)	E
American alligator (*Alligator mississippiensis*)	T/SA
Shortnose sturgeon (*Accipenser brevirostrum*)	E
Plants	
Smooth coneflower (*Echinacea laevigata*)	E
Bog spice bush (*Lindera subcoriacea*)	FC2
Boykin's lobelia (*Lobelia boykinii*)	FC2
Loose watermilfoil (*Myriophyllum laxum*)	FC2
Nestronia (*Nestronia umbellula*)	FC2
Awned meadowbeauty (*Rhexia aristosa*)	FC2

Key: E = Federal endangered species.
T/SA = Threatened due to Similarity of Appearance.
FC2 = Under review (a candidate species) for listing by the Federal government.

listed Category 2 species, and several state listed species (Knox and Sharitz 1990). At present, the SRS is implementing strategies for the protection of these species.

F- and H-Areas and the representative host site contain no habitat suitable for any of the Federally listed threatened or endangered species found on the SRS. The Southern bald eagle and the wood stork feed and nest near wetlands, streams, and reservoirs, and thus would not be attracted to the host site, a densely forested upland area. Shortnose sturgeon, typically residents of large coastal rivers and estuaries, have never been collected in Fourmile Branch or any of the tributaries of the Savannah River that drain the SRS.

Red-cockaded woodpeckers prefer open pine forests with mature trees (older than 80 years) for foraging and nesting. The pines of the undeveloped host site are 5 to 40 years old, thus red-cockaded woodpeckers probably would not forage or nest in the area.

The *Red-cockaded Woodpecker Management Standards and Guidelines, Savannah River Site* (USDA 1991b) describes the SRS management strategy for the red-cockaded woodpecker. The most significant element of this management strategy is the conversion of slash (and some loblolly) pine in a designated red-cockaded woodpecker management area to longleaf pine, with a harvest rotation of 120 years.

4.10 Noise

The major noise sources at the SRS occur primarily in developed operational areas and include various facilities, equipment, and machines (e.g., cooling towers, transformers, engines, pumps, boilers, steam vents, paging systems, construction and materials-handling equipment, and vehicles). Major noise sources outside the operational areas consist primarily of vehicles and railroad operations. Previous studies have assessed noise impacts of existing SRS operational activities (NUS 1991b; DOE 1991b; DOE 1990; DOE 1993a). These studies concluded that, because of the remote locations of the SRS operational areas, there are no known conditions associated with existing onsite noise sources that adversely affect individuals at offsite locations. Some disturbance of wildlife activities might occur on the SRS as a result of operational and construction activities.

Existing SRS-related noise sources of importance to the public are those resulting from the transportation of people and materials to and from the Site. These sources include trucks, private vehicles, helicopters, and freight trains. In addition, a portion of the air cargo and business travel using commercial air transport through the airports at Augusta, Georgia, and Columbia, South Carolina, are attributable to SRS operations.

The States of Georgia and South Carolina and the counties in which the SRS is located have not established any regulations that specify acceptable community noise levels with the exception of Aiken County. A provision of the Aiken County Nuisance Ordinance limits daytime and nighttime noise by frequency band (Aiken County 1991).

During a normal week in 1995, about 20,000 employees are likely to travel to the SRS each day in private vehicles from surrounding communities. Both government-owned and private trucks pick up and deliver materials at the Site. Most private vehicles and trucks traveling to and from the Site each day use South Carolina Highways (SC) 125 and 19. The contribution of SRS operations to traffic volumes along SC 125 and SC 19, especially during peak traffic periods, affects noise levels through the towns of New Ellenton and Jackson and the City of Aiken.

Noise measurements taken during 1989 and 1990 along SC 125 in the Town of Jackson at a point about 15 meters (50 feet) from the roadway indicate that the 1-hour equivalent sound level from traffic ranged from 48 to 72 decibels (A-weighted). The estimated day/night average sound level along this route was 66 decibels for summer and 69 decibels for winter. Similarly, noise measurements along SC 19 in the town of New Ellenton at a point about 15 meters (50 feet) from the roadway indicate that the 1-hour equivalent sound level from traffic ranged from 53 to 71 decibels. The estimated day/night average sound level along this route was 68 decibels for summer and 67 decibels for winter (NUS 1990). Employment at the SRS has increased slightly since 1989, potentially causing small increases in traffic noise, especially during peak traffic periods (approximately between 6:30 and 8:30 a.m. and between 3:30 and 5:30 p.m., corresponding to the major shift changes). Because some residences and at least two schools are within 100 to 200 feet of these routes, some annoyance to members of the public residing along these highways might occur based on the relationship between the day/night average sound level and the "percent highly annoyed" (Schultz 1978; Fidell et al. 1989; FICON 1992).

Noise sources from rail transport include diesel engines, wheel-track contact, and whistle-warnings at rail crossings.

4.11 Traffic and Transportation

4.11.1 Regional Infrastructure

The SRS is surrounded by a system of Interstate highways, U.S. highways, state highways, and railroads. The regional transportation networks service the four South Carolina counties (Aiken, Allendale, Bamberg, and Barnwell) and two Georgia counties (Columbia and Richmond) that generate about 90 percent of SRS commuter traffic (HNUS 1992a). Two major railroads - CSX Transportation and Norfolk Southern Corporation - also serve the SRS vicinity. Although barge traffic is possible on the Savannah River, neither the SRS nor commercial shippers normally use barges. Figure 4-13 shows the regional transportation infrastructure.

4.11.1.1 Regional Roads. Two Interstate highways serve the SRS area. Interstate 20 (I-20) provides a primary east-west corridor and I-520 links I-20 with parts of Augusta, Georgia. U.S. Highways 1 and 25 are principal north-south routes and U.S. 78 provides east-west connections. Several other highways - U.S. 221, U.S. 301, U.S. 321, and U.S. 601 - provide additional transport routes in the region.

Several state routes provide direct access to the SRS. Running northwest/southeast is SC 125. Access to the Site is provided from the north by SC 19, from the northeast by SC 39, and from the east by SC 64.

U.S. 278 bisects the northern part of the SRS and is available to public access without restriction. The SRS maintains barricades at site entries and exits on SC 125 to control public access if necessary, although it is generally open to unrestricted public travel. The public also has direct access to Site Road 1. All other site roads have restricted access.

4.11.1.2 Regional Railroads. Norfolk Southern serves Augusta and Savannah, Georgia, as well as Columbia and Charleston, South Carolina. CSX serves the same locations and the SRS.

4.11.2 SRS Infrastructure

The SRS transportation infrastructure consists of more than 143 miles (230 kilometers) of primary roads, 1,200 miles (1,931 kilometers) of unpaved secondary roads, and 103 kilometers (64 miles) of railroad track (WSRC 1993b). These roads and railroads provide connections among the various SRS facilities and to offsite transportation linkages. Figure 4-14 shows the SRS network of primary roadways and access points. Figure 4-15 shows the SRS railway system.

4.11.2.1 SRS Roads. Two major public highways traverse the Site: SC 125 and U.S. 278. SC 125 connects Allendale, South Carolina, to Augusta, Georgia, by crossing the Site in a northwest-to-southeast direction. U.S. 278 also connects Augusta and Allendale, but its route approximately follows the northern and eastern SRS boundaries.

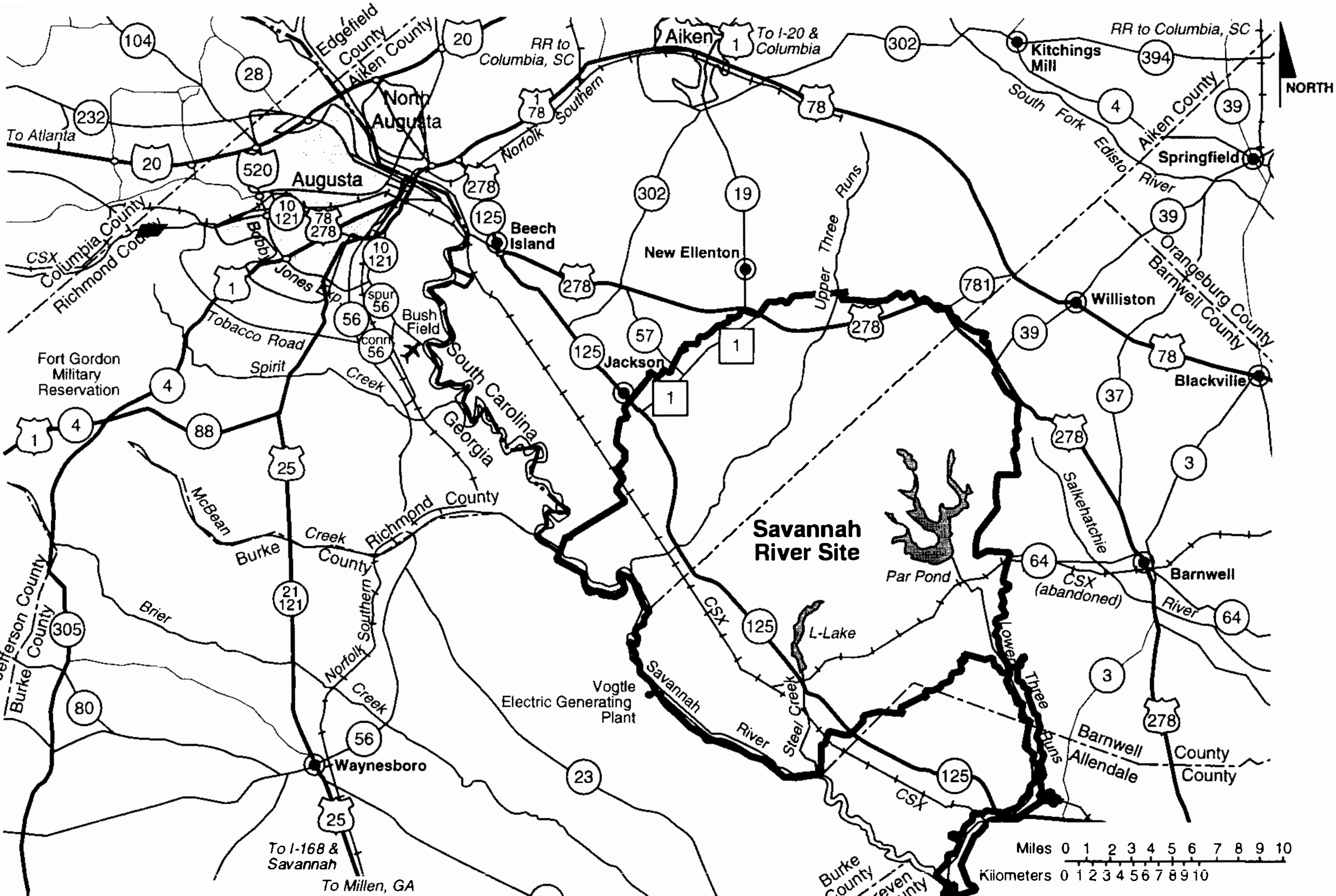


Figure 4-13. Regional transportation infrastructure.

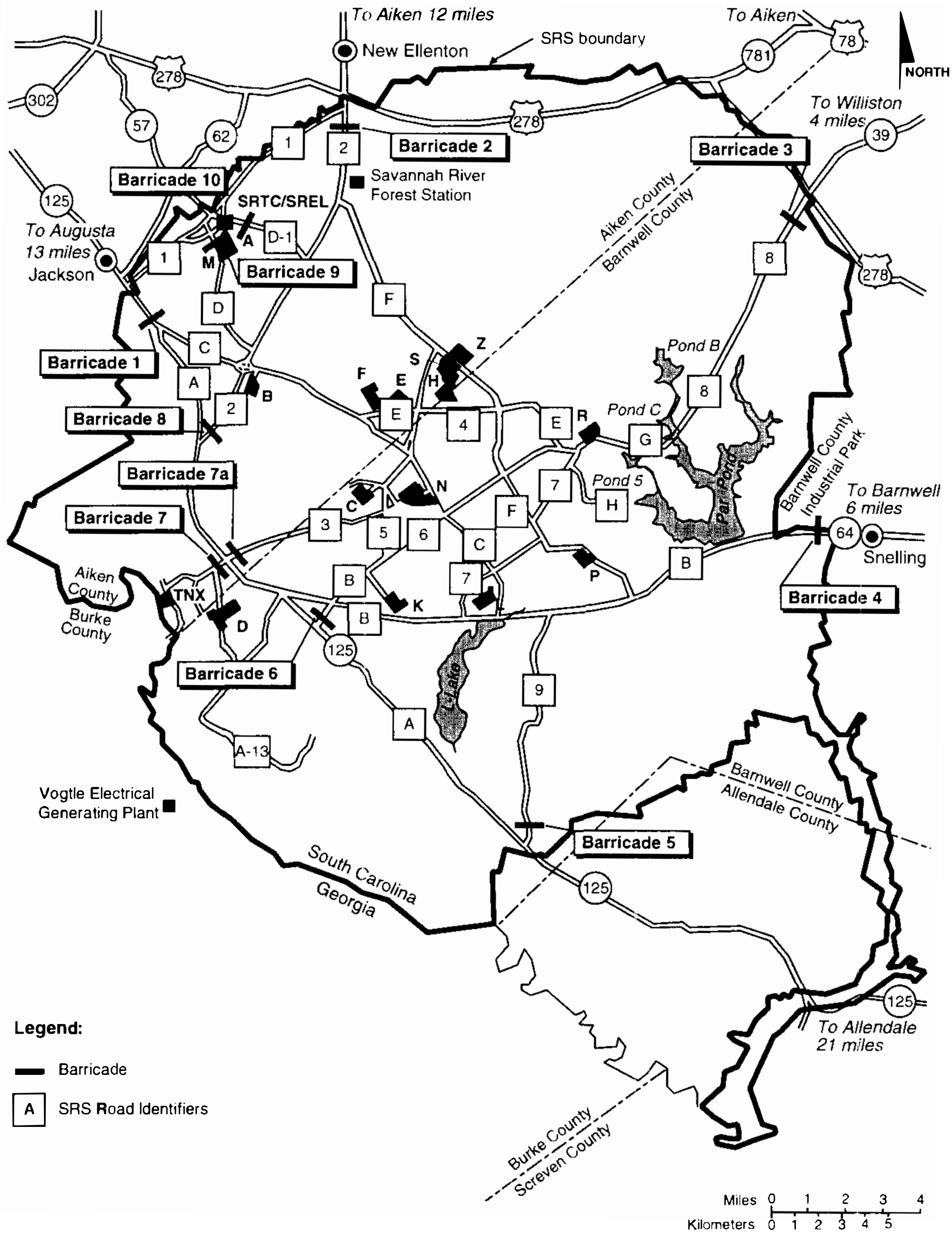


Figure 4-14. Major SRS roads and access points.

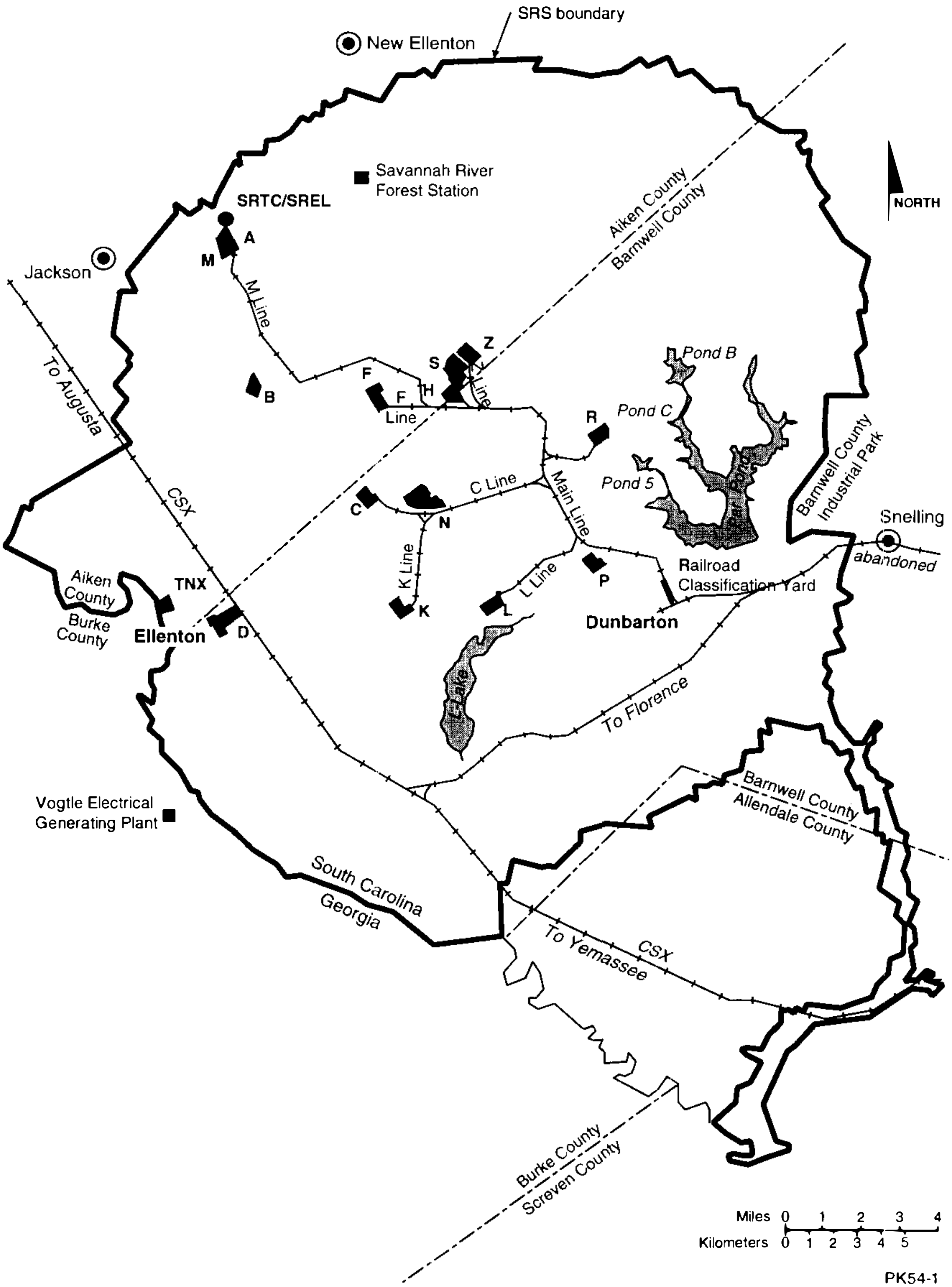


Figure 4-15. SRS railroad lines.

Ten barricades around the Site limit access from public roads. Five barricades limit SRS access from SC 125; three limit access from SC 19, SC 39, and SC 64; and two limit access from the public areas of the administrative complex near the northern SRS boundary (A-Area).

In general, the primary SRS roadways are in good condition and are smooth and free from potholes. Typically, wide, firm shoulders border roads that are either straight or have wide gradual turns. Intersections are well marked for both traffic and safety identification and are sufficiently cleared of trees and brush that might obstruct a driver's view of oncoming traffic. Railings along the side of the roadways offer protection at appropriate locations from dropoffs or other hazards. In general, the roadways are lighted only at gate areas and near major facilities. The SRS has two overpasses, one at the cloverleaf intersection of Roads 2 and C, and the other where SC 125 overpasses the CSX railroad tracks in the southern part of the Site. The 60 bridges on the Site have been inspected and evaluated for safe loading, with some bridges rated as high as 200 tons (181 metric tons) under controlled conditions. The steepest roadway gradient is on Road C at the east bank of Upper Three Runs Creek, where the road drops more than 100 feet (30 meters) in about 0.25 miles (0.4 kilometer). At the base of the dropoff is a bridge over the creek and an immediate turn in the road. This area presents a relatively hazardous roadway condition.

In general, heavy traffic occurs early in the morning and late in the afternoon when workers from surrounding communities commute to and from the Site. During working hours, official vehicles and logging trucks constitute most of the traffic. At any time, as many as 60 logging trucks, which can impede traffic, might be operating on the Site, with an annual average of about 25 trucks per day. Table 4-16 provides data on traffic counts for various roads and access points around the SRS.

4.11.2.2 SRS Railroads. Railroads on the Site include both CSX tracks and SRS rolling stock and tracks. Two routes of the CSX distribution system run through the Site: a line between Florence, South Carolina, and Augusta, Georgia, and a line between Yemassee, South Carolina, and Augusta, Georgia. The two lines join on the Site just south of L-Lake (Figure 4-15). Early in 1989 CSX discontinued service on the line from the SRS junction to Florence.

The 64 miles (103 kilometers) of SRS railroads are well maintained. The rails and crossties are in good condition, and the track lines are clear of vegetation and debris. Significant clear areas border the tracks on both sides. Intersections of railroads and roadways are marked by railroad crossing signs with lights where appropriate.

Table 4-16. SRS traffic counts - major roads.[a]

Measurement point	Date	Direction	Day Total	Peak[b]	Peak time[c]	Average speed (mph)[d]
Road 2 between Roads C and D	2-23-93	East	3,031	800	1530	47
	4-21-93	West	3,075	864	0630	NA[e]
Road 4 between Roads E and C	12-9-92	East	1,624	352	1530	NA
	12-9-92	West	1,553	306	0615	NA
Road 8 at Pond C	2-23-92	East	634	274	1530	58
	2-23-92	West	662	331	0615	56
Road C between landfill and Road 2	12-16-92	North	6,931	2,435	1530	53
	12-16-92	South	6,873	2,701	0630	58
Road C north of Road 7	1-20-93	North	742	288	0630	53
	1-20-93	South	763	223	1530	54
Road D	9-29-93	North	1,779	218	1500	43
	9-29-93	South	1,813	220	0845	52
Road E at E-Area	8-25-93	North	3,099	669	1530	35
	8-25-93	South	3,054	804	0630	38
Road F at Upper Three Runs Creek	2-2-93	North	3,239	1,438	1530	53
	2-2-93	South	3,192	1,483	0630	51
H-Area Exit	12-2-92	Outbound	2,181	406	1530	12

a. Source: Swygert (1993).
b. Number of vehicles in peak hour.
c. Start of peak hour.
d. mph = miles per hour; to convert to kilometers per hour multiply by 1.6093.
e. NA = data not available.

The SRS rail classification yard is east of P-Reactor. This eight-track facility sorts and redirects rail cars. Deliveries of SRS shipments occur at two onsite rail stations at the former towns of Ellenton and Dunbarton. From these stations, an SRS engine moves the railcars to the appropriate receiving facility. The Ellenton station, which is on the main Augusta-Yemassee line, is the preferred delivery point. The Dunbarton station, which is on the discontinued portion of the Augusta-Florence line, receives less use.

4.12 Occupational and Public Radiological Health and Safety

The sources of radiation exposure to individuals consist of natural background radiation from cosmic, terrestrial, and internal body sources; radiation from medical diagnostic and therapeutic practices; and radiation from manmade sources, including consumer and industrial products, nuclear facilities, and weapons test fallout.

All radiation doses discussed in this document are effective dose equivalents (i.e., organ dose equivalents weighted for biological effect and summed to yield a whole-body dose equivalent with the same risk as irradiation of individual organs) as defined by the International Commission on Radiological Protection, Publication 26 (ICRP 1977), unless specifically identified otherwise (e.g., thyroid dose, bone dose).

Natural background radiation contributes about 83 percent of the annual dose of 380 millirem received by an average member of the population within 50 miles (80 kilometers) of the Site. Based on national averages, medical exposure accounts for 14 percent of the annual dose, and the combined doses from weapons test fallout, consumer and industrial products, and air travel account for approximately 3 percent (Arnett et al. 1993).

4.12.1 Occupational Health and Safety

SRS maintains a network of air monitoring stations on and around the Site to determine the concentrations of radioactive particulates and aerosols in the air (Arnett et al. 1993). Table 4-17 lists average and maximum radionuclide particulate concentrations found in 1992 in air at the F- and H-Areas, SRS boundary, and background [100-mile (160-kilometer) radius] monitoring locations. Table 4-18 lists average and maximum concentrations of tritium in atmospheric moisture during 1992 for the F- and H-Areas, SRS boundary, and background monitoring locations.

Gamma radiation levels measured by thermoluminescent dosimeters in 1992 at the F- and H-Area fences averaged 70 and 74 millirem per year, respectively. Gamma radiation levels, including natural background (terrestrial and cosmic) radiation, measured at the Site perimeter in 1992 yielded an average dose of 35 millirem per year (Arnett et al. 1993).

Table 4-17. Radioactivity in air at the Savannah River Site and vicinity (pCi/m^3).[a]

Location	Gross Alpha	Nonvolatile Beta	SR-89,90[b]	Pu-238[b]	Pu-239[b]
F-Area					
Average	1.80×10^{-3}	1.94×10^{-2}	0.62×10^{-4}	1.26×10^{-5}	8.15×10^{-6}
Maximum	3.55×10^{-3}	5.56×10^{-2}	6.02×10^{-4}	2.64×10^{-5}	2.48×10^{-5}
H-Area					
Average	1.80×10^{-3}	1.93×10^{-2}	2.69×10^{-4}	2.03×10^{-5}	5.14×10^{-6}
Maximum	4.24×10^{-3}	5.39×10^{-2}	2.83×10^{-3}	6.03×10^{-5}	1.41×10^{-5}
Site perimeter					
Average	1.80×10^{-3}	2.30×10^{-2}	0.13×10^{-4}	0.01×10^{-7}	2.40×10^{-7}
Maximum	4.04×10^{-2}	4.95×10^{-2}	4.54×10^{-4}	2.21×10^{-6}	2.76×10^{-6}
Background (100-mile radius)					
Average	1.67×10^{-3}	1.73×10^{-2}	0.49×10^{-4}	0.72×10^{-6}	$<1.00 \times 10^{-6}$
Maximum	3.83×10^{-3}	4.37×10^{-2}	6.89×10^{-4}	1.98×10^{-5}	6.15×10^{-6}

a. Arnett et al. (1993).
b. Monthly composite.

Table 4-18. Tritium measured in air at the Savannah River Site (pCi/cc).[a]

Location	Average	Maximum
F-Area	8.67×10^{-5}	2.98×10^{-4}
H-Area	0.99×10^{-3}	6.77×10^{-3}
Site boundary	2.65×10^{-5}	1.03×10^{-4}
Background (100-mile radius)	8.32×10^{-6}	1.08×10^{-5}

a. Arnett (1993).

Soil samples from uncultivated areas provide a measure of the quantity of particulate radioactivity deposited from the atmosphere. Table 4-19 lists maximum measurements of radionuclides in the soil for 1992 at F- and H-Areas, SRS boundary, and background [100-mile (160-kilometer)-radius] monitoring locations. The SRS measured elevated concentrations of plutonium-238 and plutonium-239 around F- and H-Areas, reflecting releases from these areas. From 1955 through 1992, total atmospheric plutonium releases from the F- and H-Areas were approximately 0.7 curie of plutonium-238 and 3 curies of plutonium-239 (Arnett et al. 1992; 1993).

The SRS workers investigated for purposes of assessing occupational radiation exposures belong to the group of involved workers assigned to F- and H-Area facilities. The investigation selected these facilities because they process materials with radiological characteristics similar to the materials being

Table 4-19. Maximum radioactivity concentrations in soil at the Savannah River Site (pCi/g).[a]

Location	Sr-90	Cs-137	Pu-238	Pu-239
F-Area	2.16×10^{-2}	7.19×10^{-1}	4.03×10^{-1}	5.31×10^{-1}
H-Area	2.89×10^{-2}	8.22×10^{-1}	2.13×10^{-2}	5.54×10^{-2}
Site perimeter	(b)	4.84×10^{-1}	2.19×10^{-3}	1.36×10^{-2}
Background (100-mile radius)	1.46×10^{-2}	(b)	2.34×10^{-4}	1.93×10^{-2}

a. Arnett et al. (1992).
b. None detected.

analyzed in this EIS. The dosimetry results for these two involved worker groups are most useful because they depict occupational impacts that are directly relevant to each alternative. The investigation selected two dosimetry periods of record for this analysis: 1983 - 1987 and 1993. The earlier 5-year period included times when materials processing was occurring at a rate that was accelerated in comparison with recent years. The later period includes processing rates that better reflect near-term DOE mission initiatives.

Tables 4-20 and 4-21 list the involved worker dosimetry data for 1983 - 1987 and 1993, respectively. This analysis adapted these data from monitoring data statistics (Matheny 1994a; Matheny 1994b) for operations, maintenance, laboratory, and health protection personnel assigned to the F- and H-Area Canyons and the associated B-Line facilities. The calculated incidences of excess fatal cancer attributable to each facility's collective worker dose are approximately 0.11 and 0.037 for the earlier and later time periods, respectively. Similarly, the highest calculated excess fatal cancer probabilities attributable to average individual worker doses are approximately 0.0003 and 0.0001, respectively. The analysis estimated these health effects using risk coefficients adopted by DOE (DOE 1993).

4.12.2 Public Health and Safety

Table 4-22 summarizes the major sources of exposure for the population within 50 miles (80 kilometers) of the SRS and for the Savannah River water-consuming population in Beaufort and Jasper Counties, South Carolina, and Port Wentworth, Georgia. Most of the sources, such as natural background dose and medical dose, are independent of the presence of the SRS.

Atmospheric releases of radioactive material to the environment from SRS operations from 1990 to 1992 resulted in an average dose of approximately 0.02 millirem per year to individuals in the 50-mile

Table 4-20. Annual involved worker doses, 1983 - 1987.

Facility	Average Worker Dose (rem)	Total Collective Worker Dose (person-rem)
H-Canyon	0.41	36.28
HB-Line	0.49	21.84
F-Canyon	0.48	87.25
FB-Line	0.74	124.68
Facilities Average	0.53	NA
Facilities Total	NA	270.05

NA = Not applicable.

Table 4-21. Annual involved worker doses, 1993.

Facility	Average Worker Dose (rem)	Total Collective Worker Dose (person-rem)
H-Canyon	0.17	11.07
HB-Line	0.24	21.97
F-Canyon	0.22	9.16
FB-Line	0.24	51.16
Facilities Average	0.22	NA
Facilities Total	NA	93.36

NA = Not applicable.

Table 4-22. Major sources of radiation exposure to the public in the vicinity of the Savannah River Site.[a]

Source of Exposure	Dose to average individual (mrem/yr)	Percentage of exposure
Natural background radiation	315	83
Medical radiation	54	14
Consumer and industrial products, fallout, air travel	10	3
Savannah River Site operations	0.22	0.06
Grand Total	380	100

a. Arnett et al. (1993).

(80-kilometer)-radius population. The collective effective dose equivalent due to atmospheric releases from 1992 SRS operations to the population of 620,100 within 50 miles (80 kilometers) was approximately 6.4 person-rem per year. Atmospheric releases of tritium accounted for more than

90 percent of the offsite population dose; tritium is the only radionuclide of SRS origin that is routinely detected in offsite air (Cummins et al. 1991; Arnett et al. 1992, 1993). Table 4-23 lists average annual atmospheric tritium concentrations in the vicinity of SRS for the three years ending in 1992.

Table 4-23. Average atmospheric tritium concentrations in the vicinity of the Savannah River Site (pCi/m^3).[a]

Location	1992	1991	1990
Onsite	340	250	430
Site perimeter	27	21	32
25-mile radius	11	11	12
100-mile radius	8.3	8.5	8.8

a. Arnett et al. (1993).

From 1990 to 1992, the calculated maximum individual average annual dose from atmospheric releases to a hypothetical individual residing at the SRS boundary was 0.12 millirem (Cummins et al. 1991; Arnett et al. 1992, 1993).

In general, liquid releases of tritium account for more than 99 percent of the total radioactivity introduced into the Savannah River from SRS activities (Arnett et al. 1993). The calculated average annual dose to the maximally exposed individual resulting from liquid releases from 1990 to 1992 was 0.21 millirem (Cummins et al. 1991; Arnett et al. 1992; 1993). From 1990 to 1992 liquid releases of radioactive material to the environment from SRS operations resulted in an average dose of 0.04 millirem per year and 0.05 millirem per year to downstream consumers of drinking water from the Beaufort-Jasper and Port Wentworth water treatment plants, respectively. These doses to the current Beaufort-Jasper river-water-consuming population of about 51,000 and the current Port Wentworth river-water-consuming population of about 20,000 would yield a collective effective dose equivalent to these populations of approximately 3 person-rem per year (Cummins et al. 1991; Arnett et al. 1992, 1993).

The SRS analyzes samples from other environmental media that onsite releases might affect and that might provide a pathway for radiation exposure to the public and Site employees; these include samples of milk, food products, drinking water, wildlife, rainwater, soil, sediment, and vegetation. The 1992 SRS Environmental Report (Arnett et al. 1993) describes the sampling program, monitoring locations, and monitoring results for each of these media.

Major nuclear facilities within 50 miles (80 kilometers) of the SRS include a low-level waste burial site operated by Chem-Nuclear Systems, Inc., near the eastern SRS boundary in Barnwell, South Carolina, and the Georgia Power Company Alvin W. Vogtle Electric Generating Plant, directly across the Savannah River from the SRS. Plant Vogtle began commercial operation in 1987, and its releases are controlled to meet U.S. Nuclear Regulatory Commission requirements.

4.13 Utilities and Energy

This section describes SRS electricity consumption, water consumption, fuel usage, and domestic and industrial wastewater treatment. Table 4-24 contains information on the current status of these items at SRS.

Table 4-24. Current capacities and usage of utilities and energy at SRS.

ELECTRICITY	
Consumption	659,000 megawatt hours per year
Load	75 megavolt-amperes
Peak Demand	130 megavolt-amperes
Capacity	340 megavolt-amperes
WATER	
Groundwater usage	12,490 million liters (3.3 billion gallons) per year
Surface water usage (cooling)	75,700 million liters (20 billion gallons) per year
FUEL	
Oil	28.4 million liters (7.5 million gallons) per year
Coal	210,000 metric tons (230,000 tons) per year
Gasoline	4.7 million liters (1.24 million gallons) per year
WASTEWATER	
Domestic capacity	3.97 million liters (1.05 million gallons) per day
Domestic load	1.89 million liters (0.50 million gallons) per day
Industrial capacity[a,b]	1.64 million liters (433,244 gallons) per day
Industrial load[a]	44,000 liters (11,580 gallons) per day

a. F/H Effluent Treatment Facility only.
b. Design capacity; permitted capacity is about 67 percent of this value.

4.13.1 Electricity

The SRS purchases electric power from the South Carolina Electric and Gas Company (SCE&G) through three purchased power-line interconnects to the SRS transmission grid. The recent total annual power consumption for the SRS was approximately 659,000 megawatt-hours. The average load was 75 megavolt-amperes and the peak demand was about 130 megavolt-amperes. South Carolina Electric and Gas sources can supply as much as 340 megavolt-amperes to the SRS grid with existing direct connections. The SRS generating station in D-Area can produce an additional 80 megavolt-amperes capacity, although that plant currently produces only process steam. The SRS transmission grid that would provide power to any spent nuclear fuel facilities consists of more than 145 kilometers (90 miles) of 115-kilovolt lines, four switching stations, and 15 substations. Electric service to all major production areas provides parallel redundant capacity to ensure maximum availability and reliability (WSRC 1993c).

4.13.2 Water Consumption

Groundwater from a deep confined aquifer supplies domestic and process water for the SRS through approximately 100 production wells. The aquifer system sustains single well yields of about 10.2 million liters (2.7 million gallons) per day. Current usage from this source is about 14.0 billion liters (3.7 billion gallons) per year (DOE 1990). The SRS withdraws cooling water for its facilities from the Savannah River at an annual rate of about 75.7 billion liters (20 billion gallons) (WSRC 1993c).

4.13.3 Fuel Consumption

Fuels consumed at SRS include oil, coal, and gasoline. SRS facilities and equipment burn approximately 28.4 million liters (7.5 million gallons) of oil each year. This total includes diesel fuel, No. 6 oil, and No. 2 oil. The SRS burns coal and some waste oils in the D-Area powerhouse to produce steam for Site facilities. Current coal usage is about 208,655 metric tons (230,000 tons) per year. SRS vehicles use approximately 4.7 million liters (1.24 million gallons) of gasoline annually. Under the provisions of the Energy Policy Act of 1992, natural gas will replace gasoline on the SRS within the next 10 years. At that time, SRS usage of natural gas would be approximately 12.2 million cubic meters (429 million cubic feet) per year. At present, the SRS consumes no natural gas (WSRC 1993c).

4.13.4 Wastewater Treatment

By 1995, the SRS Centralized Sanitary Wastewater Treatment Facility will process most of the domestic effluent on the Site. This centrally located facility has a design capacity of 4 million liters

(1.05 million gallons) per day. Once operational, the plant will use about 50 percent of this capacity. In addition, five smaller sanitary treatment plants serve more remote areas of the Site. Facilities for spent nuclear fuel management would use the centralized facility.

The F/H Effluent Treatment Facility (ETF), which decontaminates routine process effluents and accidental radioactive releases from operations, treats industrial wastewater in the F- and H-Areas, where the spent fuel management activities would occur.

Effluent Treatment Facility process operations performed on the waste liquids include neutralization (adjusts pH), submicron filtration (removes suspended solids), activated carbon absorption (removes dissolved organic chemicals), reverse osmosis membrane deionization (removes salts), ion exchange (removes heavy metals), and evaporation (separates radionuclides from aqueous condensate). This facility releases two different streams. The treated water stream is sampled and analyzed to ensure that it meets discharge requirements and then is released to Upper Three Runs Creek via a permitted outfall. The waste concentrate (i.e., bottoms from the evaporator process) is transferred to the H-Area waste tank farm for treatment and disposal in the Z-Area Saltstone facility.

The design capacity for the Effluent Treatment Facility is approximately 600 million liters (158 million gallons) per year. The maximum permitted treatment capacity is about 400 million liters (105.7 million gallons) per year. Under normal operating conditions, the facility treats more than 16,000 cubic meters (26 million gallons) of liquid waste per year (WSRC 1993d).

The influent water load to processes discharging to the permitted outfall includes as much as 205 million liters (54 million gallons) per year of F-Area Canyon process wastewater, 120 million liters (32 million gallons) per year of H-Area Canyon process wastewater, 34 million liters (9 million gallons) per year from the F-Area collection and retention basins, 34 million liters (9 million gallons) per year from the H-Area collection and retention basins, 68 million liters (18 million gallons) per year of Effluent Treatment Facility acid, caustic, flush and rinse water, and similar wastewater from other SRS facilities.

4.14 Materials and Waste Management

The historic national defense mission of the SRS has resulted in the generation of high-level radioactive waste, transuranic waste, low-level radioactive waste (low-activity and intermediate-level), hazardous waste, mixed waste (radioactive and hazardous combined), and sanitary waste (nonhazardous, nonradioactive solid waste). This section discusses the treatment, storage, and disposal of waste at the SRS. Section 4.13 discusses domestic and industrial wastewater treatment.

DOE is preparing an environmental impact statement on Waste Management at the Savannah River Site (DOE 1995). The purpose of the EIS is to provide a basis for DOE to select a sitewide strategic approach to managing present and future SRS waste generated as a result of ongoing operations, environmental restoration activities, transition from nuclear production to other missions, and decontamination and decommissioning programs. The Waste Management EIS will support project-level decisions on the operation of specific treatment, storage, and disposal facilities within the near term (10 years or less). In addition, the EIS will provide a baseline for analyses of future waste management activities and a basis for the evaluation of the specific waste management alternatives. The Waste Management EIS will not include management of spent nuclear fuel which is addressed in this document.

DOE treats and stores waste generated from onsite operations in waste management facilities located primarily in E-, F-, H-, N-, S-, and Z-Areas (Figure 4-16). These facilities include the F- and H-Area Effluent Treatment Facility, the High-Level Waste Tank Farms, and the Solid Waste Disposal Facility. The Defense Waste Processing Facility is nearly operational and the Consolidated Incineration Facility is under construction. The SRS places sanitary and inert waste in the Interim Sanitary Landfill and the Burma Road Landfill, respectively.

DOE continues to reduce the amount of waste generated and disposed of at the SRS through waste minimization and treatment programs. DOE accomplishes waste minimization by reducing the volume, toxicity, or mobility of waste before storing or disposing of it. These activities also include more intensive surveying, waste segregation, and use of administrative and engineering controls.

The waste that DOE presently stores on the SRS includes high-level, transuranic, hazardous, mixed waste and some low-level waste. The Site stores high-level waste in underground storage tanks that have received South Carolina Department of Health and Environmental Control industrial wastewater permits, and manages them in accordance with Clean Water Act, Resource Conservation and Recovery Act, and DOE requirements. The SRS stores transuranic mixed waste on interim-status storage pads in accordance with South Carolina Department of Health and Environmental Control requirements and DOE Orders. Hazardous and mixed waste is placed in permitted or interim-status

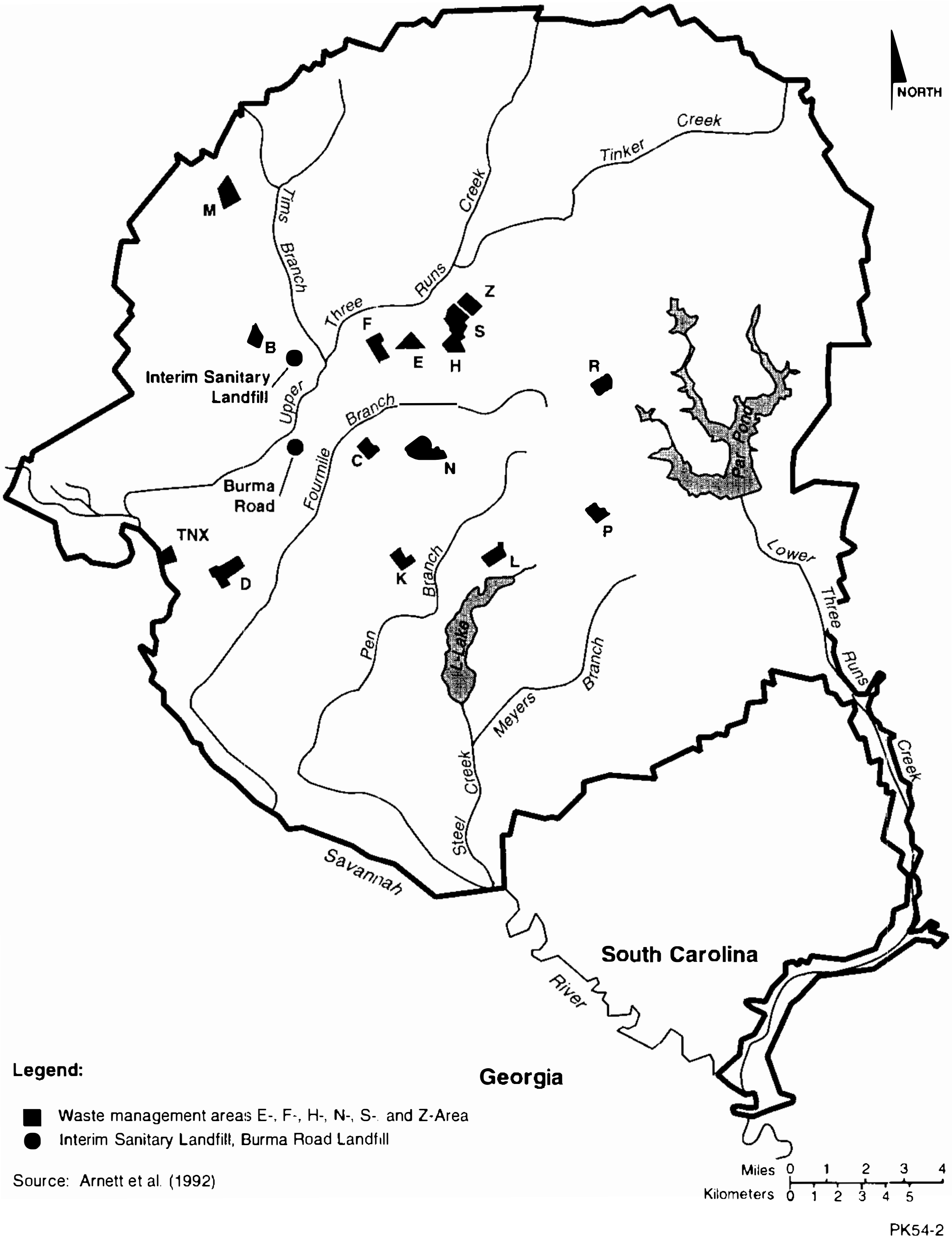


Figure 4-16. Waste management facilities at the Savannah River Site.

storage in the Hazardous Waste Storage Facilities (both buildings and pads) and in the mixed waste storage buildings.

Figure 4-17 shows the high-level liquid waste management process at the SRS. Figure 4-18 shows the process for handling all other forms of solid waste at the Site.

Table 4-25 is a forecast of annual waste generation for all waste forms except sanitary and high-level waste (WSRC 1994c). The volumes listed do not include waste related to decontamination and decommissioning because DOE has not yet completed the planning of these activities. Section 5.14 discusses potential consequences of spent nuclear fuel activities as they relate to the alternative interim storage and treatment scenarios.

4.14.1 High-Level Waste

The SRS generated high-level waste from the recovery of nuclear materials from spent fuel and target processing in the F- and H-Areas. It is stored in 50 underground tanks. These tanks also store other radioactive waste effluents (primarily low-level radioactive waste such as aqueous process waste, including purge water from storage basins for irradiated reactor fuel or fuel elements). The high-level waste is stored to permit the decay of short-lived radionuclides and allow separation of solids (sludge) from soluble waste. Evaporators concentrate soluble waste to reduce original volumes and to immobilize it as crystallized salt by successive evaporations of the liquid supernate. The SRS treats the evaporator overheads in cesium removal columns before transferring them to the F- and H-Area Effluent Treatment Facility. The SRS processes the sludge and salt to prepare them for vitrification at the Defense Waste Processing Facility (high-level waste), when it becomes operational, or stabilization at the Z-Area Saltstone Facility (low-level waste). DOE has prepared a Supplemental EIS related to Defense Waste Processing Facility operations (DOE 1994d).

By December 31, 1991, DOE had stored approximately 127.9 million liters (33.8 million gallons) of high-level radioactive waste on the Site. Estimates of current tank capacity and high-level waste forecasts should be available in 1995. In general, however, due to a number of factors, the most important of which has been the extended outage of the evaporators, the estimated inventory of waste in the high-level tanks is greater than 90 percent of existing capacity (WSRC 1994d). DOE is constructing a replacement high-level waste tank evaporator to augment or replace existing evaporators.

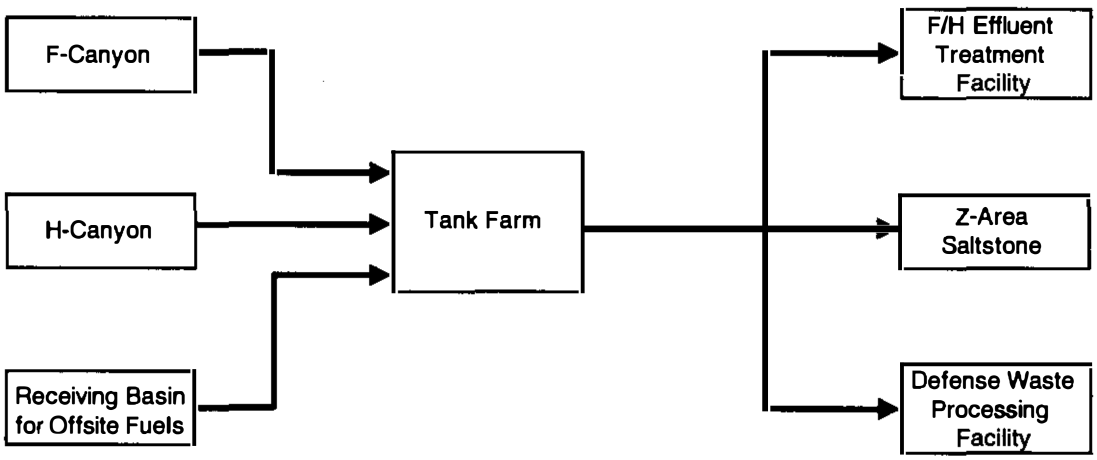
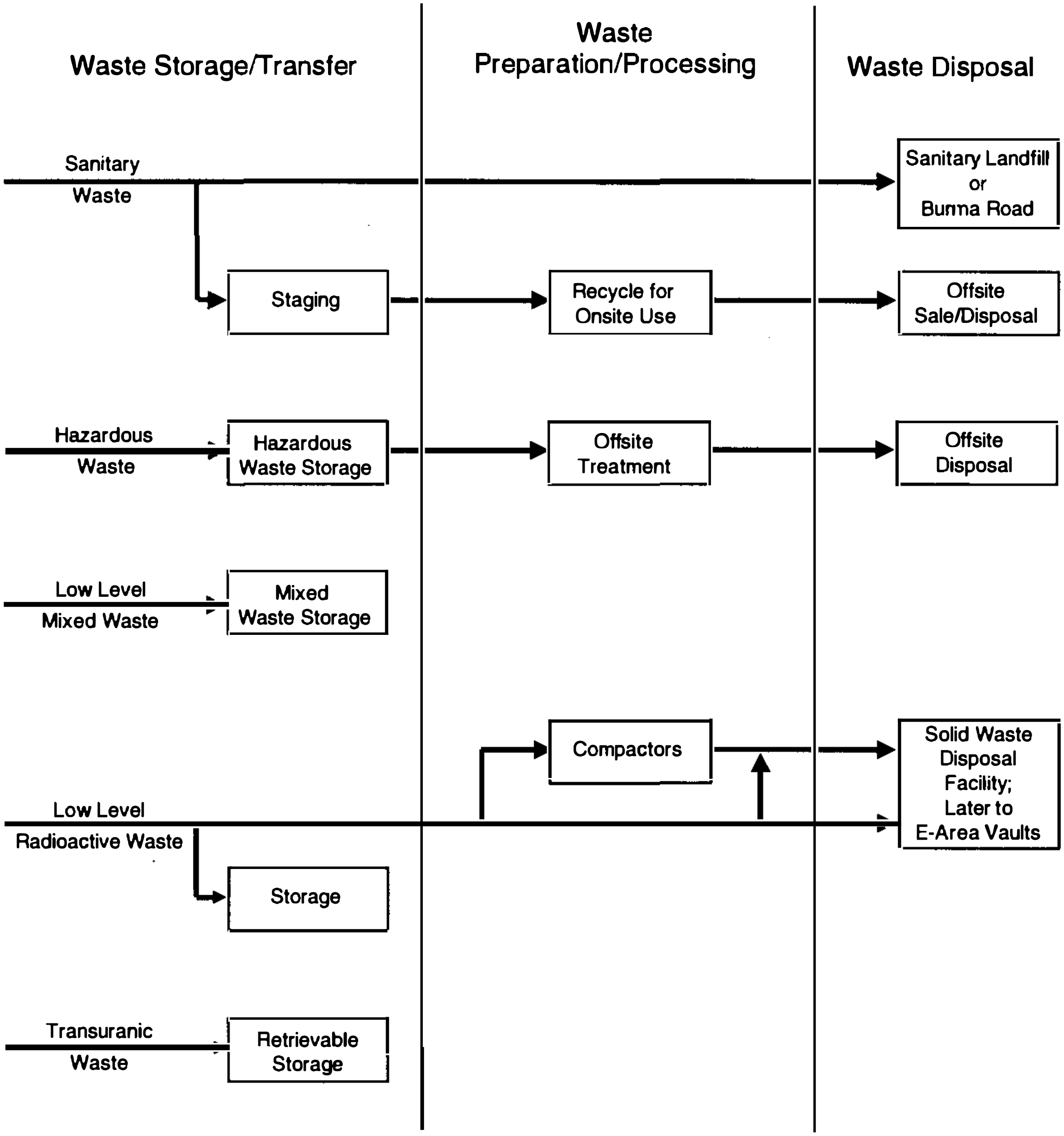


Figure 4-17. Flow diagram for high-level radioactive waste handling at the Savannah River Site.



Source: WSRC (1994e)

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Figure 4-18. Flow diagram for waste handling at the Savannah River Site.

Table 4-25. Average annual waste generation forecast for Savannah River Site (cubic meters).[a,b]

Waste Type	FY94	FY95	FY96
Transuranic	670	860	760
Low-Level			
Low-Activity	21,350	17,680	17,970
Intermediate-Level	940	580	740
Hazardous	140	130	100
Mixed	120	130	110

a. Source: WSRC (1994c).
b. To convert cubic meters to cubic feet, multiply by 35.314.

4.14.2 Transuranic Waste

At present, DOE uses three methods of retrievable storage for transuranic waste at SRS, based on the time of generation. Transuranic waste generated before 1974 is buried in approximately 120 belowgrade concrete culverts in the Solid Waste Disposal Facility. Transuranic waste generated from 1974 to 1985 is stored on five concrete pads and one asphalt pad that have been covered with approximately 1.2 meters (4 feet) of native soil. DOE stores waste generated since 1985 on 13 additional concrete pads that are not covered with soil. Pads 1 through 17 operate under Interim Status approved by the South Carolina Department of Health and Environmental Control. DOE uses Pads 18 through 19, which are not required to have interim status, to manage nonhazardous transuranic wastes only.

The SRS stores wastes containing 10 to 100 nanocuries per gram of transuranic material with transuranic waste until it can complete Site-specific radiological performance assessments, which will provide disposal limits for transuranic isotopes. SRS transuranic waste inventories and forecasts include both transuranic waste and the 10- to 100-nanocuries-per-gram transuranic wastes.

At the end of 1993, the SRS had approximately 9,900 cubic meters (350,000 cubic feet) of transuranic waste in storage (WSRC 1994e). Based on the 1994-to-1996 average annual generation rate forecast, the Site generates approximately 760 cubic meters (27,000 cubic feet) of transuranic waste annually. Transuranic mixed waste (transuranic and hazardous combined) accounts for approximately 110 cubic meters (3,900 cubic feet) of this volume (WSRC 1994c). DOE is evaluating available storage space for transuranic mixed waste to alleviate any storage capacity deficit.

4.14.3 Mixed Low-Level Waste

The SRS mixed waste program consists primarily of providing safe storage until treatment and disposal facilities are available. The current volume of mixed low-level waste at the SRS is 1,700 cubic meters (60,000 cubic feet) (WSRC 1994e). Based on the 1994-to-1996 average annual generation forecast, the Site generates approximately 118 cubic meters (4,170 cubic feet) of mixed low-level waste annually (WSRC 1994c). DOE is evaluating available storage space to determine when the SRS will exceed its capacity. However, DOE is constructing a Consolidated Incineration Facility in H-Area, which will treat mixed, hazardous, and low-level waste. When the incinerator is operational, existing inventory will be reduced and more storage capacity will become available.

4.14.4 Low-Level Waste

The SRS packages low-level waste for disposal on the Site in accordance with the waste category and its estimated surface dose rate. The Site places low-activity waste in carbon steel boxes and deposits it in an Engineered Low-Level Trench (ELLT). The trenches are several acres in size by 6 meters (20 feet) deep and have sloped sides and floor, allowing drainage to a collection sump. When the trenches are full, DOE backfills and covers them with at least 1.8 meters (6 feet) of soil. The Site packages intermediate-level wastes according to the waste form and disposes of them in slit trenches. DOE will store long-lived wastes, such as resins, until the Long-Lived Waste Storage Building, currently under construction, becomes operational. This building will provide storage until DOE develops treatment and disposal technologies.

The SRS is developing a new disposal facility, known as the E-Area Vault (EAV). This facility will include vaults for low-activity waste, intermediate-level non-tritium waste, and intermediate-level tritium waste.

Based on the 1994-to-1996 average annual generation forecast, the Site generates approximately 19,000 cubic meters (671,400 cubic feet) of low-activity waste and 750 cubic meters (26,600 cubic feet) of intermediate-level waste annually. DOE expects that the Consolidated Incineration Facility will begin operations by the second quarter of Fiscal Year 1996; this facility will have the capability of annually processing as much as 15,850 cubic meters (560,000 cubic feet) of boxed low-activity waste and approximately 186 cubic meters (6,600 cubic feet) of hazardous and mixed waste.

4.14.5 Hazardous Waste

DOE stores hazardous wastes generated at various SRS facilities in buildings in the B- and N-Areas, and on the Solid Waste Storage Pads. The Resource Conservation and Recovery Act regulates these wastes.

The inventory of hazardous waste in storage at the SRS is about 1.6 million kilograms (3.6 million pounds), occupying a volume of about 2,430 cubic meters (86,000 cubic feet) (WSRC 1994e). Based on the 1994-to-1996 average annual generation rate forecast, the Site generates approximately 124 cubic meters (4,370 cubic feet) of hazardous waste annually (WSRC 1994c).

4.14.6 Sanitary Waste

The SRS disposes of most of its solid sanitary waste in onsite landfills, the most recent of which began operation in 1985. Current disposal operations include the Interim Sanitary Landfill. About 30 trucks per work day arrive at this facility carrying approximately 18,125 kilograms (40,000 pounds) of waste that, after compaction, occupies approximately 115 cubic meters (150 cubic yards) of landfill space. The recent implementation of SRS paper and aluminum can recycling programs and disposal of office waste off the Site in a commercial landfill has increased the projected life of the landfill to the fourth quarter of 1996 (WSRC 1994e).

DOE also maintains an inert material landfill on the Site near Burma Road. This facility receives demolition and construction debris. DOE is evaluating the construction of a new SRS sanitary landfill or the use of a commercial landfill.

4.14.7 Hazardous Materials

The SRS 1993 Tier II emergency and hazardous chemical inventory lists 205 reportable hazardous substances present on the Site in excess of the 10,000-pound (4,536-kilogram) threshold quantity (WSRC 1994f). The number and the total weight of any hazardous chemicals used on the Site change daily in response to use. The annual Superfund Amendments and Reauthorization Act (SARA) reports for the SRS include listings of hazardous materials used or stored on the Site during each year.

5. ENVIRONMENTAL CONSEQUENCES

5.1 Overview

This chapter discusses the potential environmental consequences for each spent nuclear fuel management alternative described in Chapter 3. The representative host site locations, as described in Chapter 2, are the F- and H-Areas and an undeveloped site close to H-Area. These sites are representative of available areas that could support spent fuel management missions. Based on generic facility characteristics, this chapter analyzes representative consequences in terms of the environmental attributes of the potential host areas and the Savannah River Site (SRS) at large, as described in Chapter 4. Table 3-2 compares the environmental consequences of each alternative. The impacts associated with the construction and operation of a Navy Expended Core Facility are not included in this chapter, but are included in Appendix D of Volume 1 of this Environmental Impact Statement.

5.2 Land Use

Overall environmental impacts on land use by any of the alternatives would be small because the U.S. Department of Energy (DOE) would construct most new facilities in F- and H-Areas, which are already dedicated to industrial use and which previous activities have disturbed. New construction on the undeveloped representative host site near H-Area would probably be necessary only for the construction of a dry storage vault.

The Centralization Alternative (Alternative 5), under which DOE would transfer all spent nuclear fuel to the SRS, would result in the greatest changes in land use. Under this alternative, the SRS would dedicate between 70 and 100 acres (0.3 and 0.4 square kilometer) for use in spent nuclear fuel management; the exact location and size of the area affected would depend on whether DOE chose to use the wet storage, dry storage, or processing option. Of this affected area, a maximum of approximately 100 acres (0.4 square kilometer) would change from managed pine forest to industrial use.

DOE would retain under its control any lands supporting the spent nuclear fuel management program for the life of the project. No alternative would require the acquisition of public lands.

5.3 Socioeconomics

Socioeconomic consequences resulting from the implementation of any of the alternatives would relate primarily to changes in employment within the region of influence (ROI). DOE has based the analysis in the following section on estimated employment and population data for each SRS spent nuclear fuel alternative, as listed in Table 5-1. The population within the region of influence in 1995 is estimated to be approximately 462,000. The labor force will be about 257,000 persons of which about 242,000 will be employed.

DOE expects the employment level at the Site to decline from about 20,000 (in 1995) to about 15,800 (in 2004) as the SRS mission is redefined. This anticipated decline would be somewhat offset by the jobs created by the spent nuclear fuel management activities. Therefore, none of the alternatives would require additional operations employees because the SRS could fill all operational positions through the reassignment of existing workers. Consequently, this analysis addresses only employment impacts from construction activities. Given the natural variation in construction employment levels, the analysis could not accurately determine the reassignment of existing construction workers. As a result, this assessment analyzed the maximum potential impact, which assumes that all construction employment would represent new jobs that in-migrating workers would fill.

DOE estimated total employment impacts using the Regional Input-Output Modeling System that the U.S. Bureau of Economic Analysis developed for the SRS region of influence. This assessment also analyzed changes in population based on historic data that indicate that 90 percent of SRS employees live in the six-county region.

5.3.1 Potential Impacts

Table 5-1 lists direct increases in construction employment for each alternative and the corresponding change in population. As listed, potential impacts to socioeconomic resources would be smallest under Alternative I (No Action) and would be greatest under Option 5b (Centralization - Wet Storage). Therefore, Option 5b provides the bounding case for maximum potential impacts to socioeconomic resources.

Table 5-1. Direct construction employment and total population changes by alternative, 1995-2004.

Alternative	1995[a]	1996[a]	1997[a]	1998[a]	1999[a]	2000	2001	2002	2003	2004
Alternative 1-										
Employment[a]	50	50	50	50	50	50	50	50	50	50
Population	200	150	150	100	100	100	100	100	100	100
Option 2a-										
Employment	50	50	50	50	50	200	400	600	500	200
Population	200	150	150	100	100	850	1,550	2,250	2,000	750
Option 2b-										
Employment	50	50	50	50	50	200	400	600	500	200
Population	100	150	150	100	100	850	1,550	2,250	2,000	750
Option 2c-										
Employment	50	50	50	50	50	200	350	550	500	150
Population	200	150	150	100	100	700	1,350	2,050	1,850	600
Option 3a-										
Employment	50	50	50	50	50	200	400	600	500	200
Population	200	150	150	100	100	850	1,550	2,250	2,000	750
Option 3b-										
Employment	50	50	50	50	50	200	400	650	600	250
Population	200	150	150	100	100	800	1,600	2,550	2,400	900
Option 3c-										
Employment	50	50	50	50	50	200	350	550	500	150
Population	200	150	150	100	100	700	1,350	2,050	1,850	600
Option 4a-										
Employment	50	50	50	50	50	200	400	650	600	250
Population	200	150	150	100	100	800	1,600	2,550	2,400	900
Option 4b-										
Employment	50	50	50	50	50	200	400	650	600	250
Population	200	150	150	100	100	800	1,600	2,550	2,400	900
Option 4c-										
Employment	50	50	50	50	50	200	350	550	500	150
Population	200	150	150	100	100	700	1,350	2,050	1,850	600
Option 4d-										
Employment	50	50	50	50	50	300	500	700	650	250
Population	200	200	150	150	150	1,100	1,900	2,800	2,500	900
Option 4e-										
Employment	50	50	50	50	50	250	500	800	800	300
Population	200	200	150	150	150	1,000	2,000	3,200	3,000	1,100
Option 4f-										
Employment	50	50	50	50	50	200	450	650	600	200
Population	200	200	150	150	150	850	1,700	2,550	2,350	700
Option 4g-										
Employment	50	50	50	50	50	100	150	200	100	100
Population	200	150	150	100	100	250	500	700	450	300

Table 5-1. (continued).

Alternative	1995[a]	1996[a]	1997[a]	1998[a]	1999[a]	2000	2001	2002	2003	2004
Option 5a-										
Employment	50	50	50	50	50	900	1,750	2,550	2,500	2,450
Population	200	150	150	100	100	3,500	6,800	9,900	9,700	9,450
Option 5b-										
Employment	50	50	50	50	50	1,000	1,900	2,700	2,650	2,600
Population	200	150	150	100	100	3,850	7,450	10,550	10,350	10,100
Option 5c-										
Employment	50	50	50	50	50	900	1,750	2,550	2,500	2,450
Population	200	150	150	100	100	3,500	6,800	9,900	9,700	9,500
Option 5d-										
Employment	50	50	50	50	50	100	150	200	100	100
Population	200	150	150	100	100	250	500	700	450	300

a. Construction is related to renovation of reactor basin and Receiving Basin for Offsite Fuels.

Table 5-2 lists indirect employment and corresponding population changes associated with construction phase activities under Option 5b. As listed, the number of full-time construction workers required to support the implementation of this option from 1995 to 2004 would range from approximately 50 to 2,700. When added to the indirect employment of 1,600 jobs in the peak year (2002), the total employment impact in the region would be approximately 4,300 employees.

Table 5-2. Estimated increases in employment and population related to construction activities for Option 5b, from 1995 to 2004. ROI refers to the six-county region of influence.

Factor	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Direct employment	50	50	50	50	50	1,000	1,900	2,700	2,650	2,600
Secondary employment	30	30	30	30	30	600	1,100	1,600	1,550	1,500
Total employment change	80	80	80	80	80	1,600	3,000	4,300	4,200	4,100
% Change in ROI labor force	0.03	0.03	0.03	0.03	0.03	0.54	1.00	1.41	1.36	1.32
% Change in ROI employment	0.03	0.03	0.03	0.03	0.03	0.57	1.06	1.50	1.45	1.40
Population change (in region)	200	150	150	100	100	3,850	7,450	10,550	10,350	10,100
% Change in ROI population	0.04	0.03	0.03	0.02	0.02	0.81	1.56	2.21	2.16	2.11

Assuming in-migrating workers filled all jobs, the regional labor force and employment would increase by 1.4 percent and 1.5 percent, respectively. These changes would be temporary and would have no adverse impact on the region. After 2004, employment would gradually decline to a relatively constant level of about 50 jobs.

Based on historic data, approximately 90 percent of new employees would live within the six-county region of influence. Assuming each new employee represented one household with 2.72 persons per household, there would be approximately 10,550 additional people in the region during the peak year (2002). These changes would be temporary and would represent an estimated 2.2 percent increase in baseline population levels. Given this minor change in population, DOE expects potential impacts on the demand for community resources and services such as housing, schools, police, health care, and fire protection to be negligible.

Because all the other alternatives would require fewer employees, they would result in smaller changes than those listed in Table 5-2, and would have no adverse impacts on socioeconomic resources in the region of influence.

5.4 Cultural Resources

A Programmatic Memorandum of Agreement (SRARP 1989) between the DOE Savannah River Operations Office, the South Carolina State Historic Preservation Office, and the Advisory Council on Historic Preservation, ratified on August 24, 1990, is the instrument for the management of cultural resources at the SRS. DOE uses this memorandum to identify cultural resources, assess them in terms of eligibility for the National Register of Historic Places, and develop mitigation plans for affected resources in consultation with the State Historic Preservation Officer. DOE would comply with the terms of the memorandum for all activities needed to support spent nuclear fuel management actions.

The potential for adverse impacts on cultural resources would be smallest under Alternative 1 (No Action) and would be greatest under Alternative 5 (Centralization). Any facilities that DOE would construct in F- and H-Areas, north of Road E (Alternatives 1-5), would be in Sensitivity Zones 2 and 3. Section 4.4 describes these zones. The undeveloped representative host site south and east of H-Area (Alternative 5) is in Sensitivity Zone 3. Although there are no known archeological sites in the area, it has never been surveyed. Surveying being conducted near F-Area (north of Road C and west of Road 4 along Upper Three Runs Creek) has recorded some historic and prehistoric sites. However, DOE expects no impacts in F- and H-Areas due to their extensive industrial development. Until DOE has determined the precise locations of facilities connected with any of the alternatives, it cannot predict impacts on cultural resources in the undeveloped site area (Sassaman 1994). However, DOE would mitigate, through avoidance or removal, impacts to potentially significant resources that future site surveys might discover.

5.5 Aesthetic and Scenic Resources

None of the alternatives for spent nuclear fuel management at the SRS would have adverse consequences on scenic resources or aesthetics. Most new construction would be in F- or H-Area, both of which are already dedicated to industrial use. New construction on the undeveloped site, which would occur primarily under Alternative 5, would be adjacent to H-Area in an already heavily industrialized portion of the SRS. In all cases, new construction would not be visible off the Site or from public access roads on the Site. No alternative would produce emissions to the atmosphere that would be visible or would indirectly reduce visibility.

5.6 Geologic Resources

The SRS contains no unique geologic features or minerals of economic value. Therefore, DOE anticipates no impacts to geologic resources at the SRS from any of the spent nuclear fuel management alternatives.

Other sections in this chapter consider the relationships of the Site's specific geology and the region's historic and analyzed seismicity to the local environment and to SRS spent nuclear fuel-related structures and facilities. Section 5.8 discusses the consequences of analyzed seismic events on both surface-water and groundwater resources. Section 5.15 describes estimates of risk that consider both the probability of and the consequences from a wide range of seismic events, ranging from local and regional historically documented earthquakes to postulated lower probability, higher consequence events.

The accident analyses in this chapter, which DOE based on information from approved safety analysis reports for applicable facilities, address the frequency and consequences of historic earthquakes, as well as postulated less likely, but more damaging, seismic events. DOE has evaluated the consequences from seismic challenges to the facilities and structures up to 0.20g lateral ground acceleration.

5.7 Air Quality Consequences

The SRS is in compliance with both Federal and state ambient air quality standards for criteria and toxic air pollutants. As shown in the following tables, the predicted incremental air pollutant impacts would not contribute to exceeding either the National Ambient Air Quality Standards or South Carolina's Ambient Air Quality Standards.

DOE performed analyses using computer models in order to assess the potential air quality impacts of operations under each of the spent nuclear fuel management alternatives. This section describes the results of these analyses. All the concentrations discussed below are ground-level estimations based on results from the ISC2 and FDM models for nonradiological pollutants, and MAXIGASP- and POPGASP SRS-climatology-specific models for radionuclides. The analyses assume that facility operations would result in both radiological and nonradiological emissions. DOE assessed construction impacts qualitatively in relation to the land area to be disturbed under each alternative.

Nonradiological Emissions. DOE analyzed the potential incremental impacts of only those substances for which it expects releases to the atmosphere during the normal operation of spent nuclear fuel facilities. The nonradiological releases evaluated for each alternative include seven criteria pollutants and 23 toxic pollutants. DOE selected the toxic substances for analysis by comparing the anticipated chemical usage at the proposed spent nuclear fuel facilities to the list of 257 toxic air pollutants in the South Carolina Air Pollution Regulations (SCDHEC 1976). The SRS modeled potential emissions of the listed toxic chemicals that DOE anticipates would be used during spent nuclear fuel activities. The following subsections discuss the results for both criteria and toxic pollutants. Tables 5-3 and 5-4 list the estimated maximum incremental concentrations of these pollutants at the Site boundary, while Tables 5-5 and 5-6 contain the incremental rates of release.

Radiological Emissions. DOE evaluated the potential radiological releases to the atmosphere from spent fuel management at the SRS using existing Site historical operations information. Based on the actual 1993 emissions data from the Receiving Basin for Offsite Fuels (WSRC 1994d), DOE estimates that emissions from any of the wet storage options under Alternatives 1 through 4 would

Table 5-3. Estimated incremental air quality impacts at the Savannah River Site boundary from operations of spent nuclear fuel alternatives - criteria pollutants (μg/m^3).[a]

					Incremental Concentrations from Alternatives						
					No Action	Decentralization			1992/1993 Planning Basis		
Pollutant[b]	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[e]	1	2a	2b	2c	3a	3b	3c
CRITERIA POLLUTANTS (μg/m^3)											
Carbon monoxide	8-hour	10,000	818	23	<0.01	0.1	0.1	4.3	0.1	0.1	4.3
	1-hour	40,000	3,553	180	<0.01	0.8	0.8	32	0.8	0.8	32
Ozone (as VOC)	1-hour	245	N/A[d]	N/A[d]	1.6	0.3	0.3	2.6	0.3	0.3	2.6
Nitrogen oxides	Annual geometric mean	100	30	4	<0.01	0.01	<0.01	11.00	<0.01	<0.01	11.0
Particulate matter (<10μm)	Annual	50	9	3	—	—	—	<0.01	—	—	0.01
	24-hour	150	93	56	—	—	—	0.40	—	—	0.40
Total suspended particulates (TSP)	Annual	75	20	11	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Sulfur dioxide	Annual	80	18	10	—	<0.01	<0.01	0.01	<0.01	<0.01	0.01
	24-hour	365	356	185	—	0.01	0.01	0.43	0.01	0.01	0.43
	3-hour	1,300	1,210	634	—	0.05	0.05	3.2	0.05	0.05	3.2
Lead	Calendar quarter mean	1.5	<0.01	<0.01	—	—	—	—	—	—	—
Gaseous Fluorides (as HF)	1-month	0.8	0.11	0.03	—	—	—	0.02	—	—	0.02
	1-week	1.6	0.6	0.15	—	—	—	0.10	—	—	0.10
	24-hour	2.9	1.20	0.31	—	—	—	0.20	—	—	0.20
	12-hour	3.7	2.40	0.62	—	—	—	0.40	—	—	0.40

Table 5-3. (continued).

Pollutant[b]	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[e]	Incremental Concentrations from Alternatives: Regionalization A			Regionalization B			
					4a	4b	4c	4d	4e	4f	4g
CRITERIA POLLUTANTS (μg/m³)											
Carbon monoxide	8-hour	10,000	818	23	0.2	0.2	4.3	0.2	0.2	5.5	—
	1-hour	40,000	3,553	180	1.2	1.2	32	1.5	1.5	41	—
Ozone (as VOC)	1-hour	245	N/A[d]	N/A[d]	0.5	0.5	2.6	0.6	0.6	3.3	1.4
Nitrogen oxides	Annual geometric mean	100	30	4	<0.01	<0.01	11	<0.01	<0.01	14	—
Particulate matter (<10μm)	Annual	50	9	3	—	—	0.01	—	—	0.01	—
	24-hour	150	93	56	—	—	0.4	—	—	0.5	—
Total suspended particulates (TSP)	Annual	75	20	11	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	—
Sulfur dioxide	Annual	80	18	10	<0.01	<0.01	0.01	<0.01	<0.01	0.01	—
	24-hour	365	356	185	0.02	0.02	0.43	0.02	0.02	0.55	—
	3-hour	1,300	1,210	634	0.09	0.09	3.2	0.11	0.11	4.1	—
Lead	Calendar quarter mean	1.5	<0.01	<0.01	—	—	—	—	—	—	—
Gaseous Fluorides (as HF)	1-month	0.8	0.11	0.03	—	—	0.02	—	—	0.02	—
	1-week	1.6	0.6	0.15	—	—	0.10	—	—	0.13	—
	24-hour	2.9	1.20	0.31	—	—	0.20	—	—	0.25	—
	12-hour	3.7	2.40	0.62	—	—	0.40	—	—	0.51	—

Table 5-3. (continued).

	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[e]	Incremental Concentrations from Alternatives: Centralization 5a	5b	5c	5d
CRITERIA POLLUTANTS (μg/m³)								
Carbon monoxide	8-hour	10,000	818	23	1.0	1.0	5.1	—
	1-hour	40,000	3,553	180	6.7	6.7	37	—
Ozone (as VOC)	1-hour	245	N/A[d]	N/A[d]	1.4	1.4	3.1	1.4
Nitrogen oxides	Annual geometric mean	100	30	4	0.04	0.04	11.1	—
Particulate matter (<10μm)	Annual	50	9	3	—	—	0.01	—
	24-hour	150	93	56	—	—	0.40	—
Total suspended particulates (TSP)	Annual	75	20	11	<0.01	<0.01	<0.01	—
Sulfur dioxide	Annual	80	18	10	<0.01	<0.01	0.02	—
	24-hour	365	356	185	0.09	0.09	0.49	—
	3-hour	1,300	1,210	634	0.50	0.50	3.5	—
Lead	Calendar quarter mean	1.5	<0.01	<0.01	—	—	—	—
Gaseous Fluorides (as HF)	1-month	0.8	0.11	0.03	—	—	0.02	—
	1-week	1.6	0.6	0.15	—	—	0.10	—
	24-hour	2.9	1.20	0.31	—	—	0.10	—
	12-hour	3.7	2.40	0.62	—	—	0.40	—

— = No impact.

a. Maximum modeled ground-level concentration at SRS perimeter unless higher offsite concentrations are otherwise specified.

b. Major pollutants of concern regarding spent nuclear fuel management activities.

c. Most stringent Federal and state regulatory standards (CFR 1991a), (SCDHEC 1976).

d. Measurement data currently unavailable.

e. Maximum operational air pollutant emissions projected for baseline year 1995. Concentration estimates based on actual emissions from all SRS sources for calendar year 1990 plus maximum potential emissions for sources permitted through December 1992.

Table 5-4. Estimated incremental air quality impacts at the Savannah River Site boundary from operations of spent nuclear fuel alternatives - toxic pollutants ($\mu g/m^3$).[a]

					Incremental Concentrations from Alternatives						
					No Action	Decentralization			1992/1993 Planning Basis		
Pollutant[b]	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[d]	1	2a	2b	2c	3a	3b	3c
TOXIC POLLUTANTS ($\mu g/m^3$)											
Nitric acid	24-hour	125	51	6.7	—	—	—	<0.01	—	—	<0.01
1,1,1,- Trichloroethane	24-hour	9,550	81	22	<0.01	<0.01	<0.01	0.01	<0.01	<0.01	0.01
Benzene	24-hour	150	32	31	—	—	—	0.04	—	—	0.04
Ethanolamine	24-hour	200	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	0.58	0.12	—	—	—	<0.01	—	—	<0.01
Ethylene glycol	24-hour	650	0.20	0.08	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	7.5	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronapthalene	24-hour	1.0	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	200	0.21	0.07	<0.01	<0.01	<0.01	0.04	<0.01	<0.01	0.04
Manganese	24-hour	25	0.82	0.10	—	—	—	<0.01	—	—	<0.01
Methyl alcohol	24-hour	1,310	2.9	0.51	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	6.0	0.99	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	3.0	0.51	—	—	—	<0.01	—	—	<0.01
Methylene chloride	24-hour	515	10.5	1.8	—	—	—	0.02	—	—	0.02
Naphthalene	24-hour	1,250	0.01	0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	0.03	0.03	—	—	—	<0.01	—	—	<0.01
Phosphorus	24-hour	0.5	<0.001	<0.001	—	—	—	<0.001	—	—	<0.001
Sodium hydroxide	24-hour	20	0.01	0.01	—	—	—	<0.01	—	—	<0.01
Toluene	24-hour	2,000	9.3	1.6	<0.01	<0.01	<0.01	0.04	<0.01	<0.01	0.04
Trichloroethylene	24-hour	6,750	4.8	1.0	—	—	—	<0.01	—	—	<0.01
Vinyl acetate	24-hour	176	0.06	0.02	—	—	—	<0.01	—	—	<0.01
Xylene	24-hour	4,350	39	3.8	0.01	0.01	0.01	0.05	0.01	0.01	0.05

Table 5-4. (continued).

Pollutant[b]	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[d]	Incremental Concentrations from Alternatives						
					Regionalization A			Regionalization B			
					4a	4b	4c	4d	4e	4f	4g
TOXIC POLLUTANTS (μg/m^3)											
Nitric acid	24-hour	125	51	6.7	—	—	1.0	—	—	1.3	—
1,1,1,- Trichloroethane	24-hour	9,550	81	22	<0.01	<0.01	0.01	<0.01	<0.01	0.01	<0.01
Benzene	24-hour	150	32	31	—	—	0.04	—	—	0.05	—
Ethanolamine	24-hour	200	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	0.58	0.12	—	—	<0.01	—	—	<0.01	—
Ethylene glycol	24-hour	650	0.20	0.08	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	7.5	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronapthalene	24-hour	1.0	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	200	0.21	0.07	<0.01	<0.01	0.04	<0.01	<0.01	0.05	<0.01
Manganese	24-hour	25	0.82	0.10	—	—	<0.01	—	—	<0.01	—
Methyl alcohol	24-hour	1,310	2.9	0.51	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	6.0	0.99	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	3.0	0.51	—	—	<0.01	—	—	<0.01	—
Methylene chloride	24-hour	515	10.5	1.8	—	—	0.02	—	—	0.02	—
Naphthalene	24-hour	1,250	0.01	0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	0.03	0.03	—	—	<0.01	—	—	<0.01	—
Phosphorus	24-hour	0.5	<0.001	<0.001	—	—	<0.001	—	—	<0.001	—
Sodium hydroxide	24-hour	20	0.01	0.01	—	—	<0.01	—	—	<0.01	—
Toluene	24-hour	2,000	9.3	1.6	<0.01	<0.01	0.04	<0.01	<0.01	<0.05	<0.01
Trichloroethylene	24-hour	6,750	4.8	1.0	—	—	<0.01	—	—	<0.01	—
Vinyl acetate	24-hour	176	0.06	0.02	—	—	<0.01	—	—	<0.01	—
Xylene	24-hour	4,350	39	3.8	0.01	0.01	0.05	0.01	0.01	0.06	0.01

Table 5-4. (continued).

Pollutant[b]	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[d]	Incremental Concentrations from Alternatives: Centralization 5a	5b	5c	5d
TOXIC POLLUTANTS (μg/m³)								
Nitric acid	24-hour	125	51	6.7	—	—	1.0	—
1,1,1,- Trichloroethane	24-hour	9,550	81	22	<0.01	<0.01	0.01	<0.01
Benzene	24-hour	150	32	31	—	—	0.04	—
Ethanolamine	24-hour	200	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	0.58	0.12	—	—	<0.01	—
Ethylene glycol	24-hour	650	0.20	0.08	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	7.5	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronapthalene	24-hour	1.0	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	200	0.21	0.07	<0.01	<0.01	0.04	<0.01
Manganese	24-hour	25	0.82	0.10	—	—	<0.01	—
Methyl alcohol	24-hour	1,310	2.9	0.51	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	6.0	0.99	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	3.0	0.51	—	—	<0.01	—
Methylene chloride	24-hour	515	10.5	1.8	—	—	0.02	—
Naphthalene	24-hour	1,250	0.01	0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	0.03	0.03	—	—	<0.01	—
Phosphorus	24-hour	0.5	<0.001	<0.001	—	—	<0.001	—
Sodium hydroxide	24-hour	20	0.01	0.01	—	—	<0.01	—
Toluene	24-hour	2,000	9.3	1.6	<0.01	<0.01	0.04	<0.01
Trichloroethylene	24-hour	6,750	4.8	1.0	—	—	<0.01	—
Vinyl acetate	24-hour	176	0.06	0.02	—	—	<0.01	—
Xylene	24-hour	4,350	39	3.8	0.01	0.01	0.05	0.01

— No impact.
\+ Not available.
a. Maximum modeled ground-level concentration at SRS perimeter unless higher offsite concentrations are otherwise specified.
b. Major pollutants of concern regarding spent nuclear fuel.
c. Most stringent Federal and state regulatory standards (CFR 1991a), (SCDHEC 1976).
d. Maximum operational air pollutant emissions projected for baseline year 1995. Concentration estimates based on actual emissions from all SRS sources for calendar year 1990 plus maximum potential emissions for sources permitted through December 1992.

Table 5-5. Incremental air quality pollutant emission rates related to spent nuclear fuel alternatives - criteria pollutants.[a]

	Baseline		Alternatives						
			No Action	Decentralization			1992/1993 Planning Basis		
Pollutant	Maximum Design Capacity	Actual[b]	1	2a	2b	2c	3a	3b	3c
CRITERIA POLLUTANTS (TONS PER YEAR)									
NO_x	2.22×10^{4}	2.62×10^{3}	—	6.0×10^{0}	6.0×10^{0}	2.0×10^{4}	6.0×10^{0}	6.0×10^{0}	2.0×10^{4}
Particulates									
TSP	3.62×10^{3}	9.80×10^{2}	—	4.0×10^{-1}	4.0×10^{-1}	1.5×10^{1}	4.0×10^{-1}	4.0×10^{-1}	1.5×10^{1}
PM_{10}	2.66×10^{3}	4.97×10^{2}	—	2.6×10^{-1}	2.6×10^{-1}	9.3×10^{0}	2.6×10^{-1}	2.6×10^{-1}	9.3×10^{0}
CO	6.77×10^{3}	1.99×10^{2}	—	1.5×10^{0}	1.5×10^{0}	3.8×10^{1}	1.5×10^{0}	1.5×10^{0}	3.8×10^{1}
SO_2	6.42×10^{4}	6.68×10^{3}	1.6×10^{-3}	4.0×10^{-1}	4.0×10^{-1}	1.2×10^{1}	4.0×10^{-1}	4.0×10^{-1}	1.2×10^{1}
Gaseous Fluorides	2.14×10^{-2}	1.07×10^{-2}	—	—	—	2.4×10^{1}	—	—	2.4×10^{1}
Ozone (as VOC)	N/A[c]	N/A[c]	—	6.0×10^{-1}	6.0×10^{-1}	1.8×10^{-1}	6.0×10^{-1}	6.0×10^{-1}	1.8×10^{-1}
			Regionalization A				Regionalization B		
CRITERIA POLLUTANTS (TONS PER YEAR)			4a	4b	4c	4d	4e	4f	4g
NO_x	2.22×10^{4}	2.62×10^{3}	8.5×10^{0}	8.5×10^{0}	2.0×10^{4}	1.1×10^{1}	1.1×10^{1}	2.5×10^{4}	—
Particulates									
TSP	3.62×10^{3}	9.80×10^{2}	6.0×10^{-2}	6.0×10^{-2}	1.5×10^{1}	7.6×10^{-2}	7.6×10^{-2}	1.5×10^{1}	—
PM_{10}	2.66×10^{3}	4.97×10^{2}	1.45×10^{1}	1.45×10^{1}	9.3×10^{0}	1.8×10^{1}	1.8×10^{1}	9.3×10^{0}	—
CO	6.77×10^{3}	1.99×10^{2}	2.0×10^{0}	2.0×10^{0}	3.8×10^{1}	2.5×10^{0}	2.5×10^{0}	5.2×10^{1}	—
SO_2	6.42×10^{4}	6.68×10^{3}	5.5×10^{-2}	5.5×10^{-2}	1.3×10^{1}	7.6×10^{-2}	7.6×10^{-2}	1.7×10^{1}	—
Gaseous Fluorides	2.14×10^{-2}	1.07×10^{-2}	—	—	2.4×10^{1}	—	—	3.0×10^{1}	—
Ozone (as VOC)	N/A[c]	N/A[c]	8.5×10^{-1}	8.5×10^{-1}	1.8×10^{-1}	1.1×10^{0}	1.1×10^{0}	2.3×10^{-1}	—

Table 5-5. (continued).

Pollutant	Maximum Design Capacity	Actual[b]	Alternatives			
			Centralization			
CRITERIA POLLUTANTS (TONS PER YEAR)			5a	5b	5c	5d
NO_x	$2.2x10^4$	$2.6x10^3$	$5.6x10^1$	$5.6x10^1$	$2.0x10^4$	—
Particulates						
TSP	$3.62x10^3$	$9.8x10^2$	$2.1x10^0$	$2.1x10^0$	$1.8x10^1$	—
PM_{10}	$2.66x10^3$	$4.97x10^2$	$1.4x10^0$	$1.4x10^0$	$9.3x10^0$	—
CO	$6.77x10^3$	$1.99x10^2$	$2.7x10^1$	$2.7x10^1$	$6.9x10^1$	—
SO_2	$6.42x10^4$	$6.68x10^3$	$8.1x10^0$	$8.1x10^0$	$2.0x10^1$	—
Gaseous Fluorides	$2.14x10^{-2}$	$1.07x10^{-2}$			$2.4x10^1$	—
Ozone (as VOC)	N/A[c]	N/A[c]	$4.6x10^0$	$4.6x10^0$	$2.4x10^1$	—

a. Source: WSRC (1994a).

b. Maximum operational air pollutant emissions projected for baseline year 1995. Concentration estimates based on actual emissions from all SRS sources for calendar year 1990 plus maximum potential emissions for sources permitted through December 1992.

c. Emissions data currently unavailable.

— No proposed incremental emissions.

Table 5-6. Incremental air quality pollutant emission rates related to spent nuclear fuel alternatives - toxic pollutants.[a]

Pollutant	Baseline		Alternatives						
	Maximum Design Capacity	Actual[b]	No Action	Decentralization			1992/1993 Planning Basis		
			1	2a	2b	2c	3a	3b	3c
TOXIC POLLUTANTS (TONS PER YEAR)									
Nitric Acid	1.13×10^{3}	2.56×10^{0}	5.1×10^{-2}	5.1×10^{-2}	5.1×10^{-2}	1.24×10^{2}	5.1×10^{-2}	5.1×10^{-2}	1.24×10^{2}
1,1,1-Trichloroethane	8.0×10^{1}	NA[c]	—	—	—	7.02×10^{-1}	—	—	7.02×10^{-1}
Benzene	2.9×10^{1}	4.48×10^{0}	—	—	—	8.02×10^{-1}	—	—	8.02×10^{-1}
Ethanolamine	2.21×10^{-2}	5.35×10^{-3}	1.46×10^{-3}	1.46×10^{-3}	1.46×10^{-3}	1.46×10^{-3}	1.46×10^{-3}	1.46×10^{-3}	1.46×10^{-3}
Ethyl Benzene	2.56×10^{0}	1.07×10^{0}	—	—	—	8.02×10^{-4}	—	—	8.02×10^{-4}
Ethylene Glycol	6.83×10^{-1}	4.17×10^{-1}	2.25×10^{-2}	2.25×10^{-2}	2.25×10^{-2}	4.27×10^{-2}	2.25×10^{-2}	2.25×10^{-2}	4.27×10^{-2}
Formaldehyde	4.55×10^{-2}	4.8×10^{-4}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}
Glycol Ethers	4.36×10^{-3}	1.99×10^{-4}	4.06×10^{-3}	4.06×10^{-3}	4.06×10^{-3}	4.06×10^{-3}	4.06×10^{-3}	4.06×10^{-3}	4.06×10^{-3}
Hexachloronaphthalene	<0.01	NA[c]	3.65×10^{-5}	3.65×10^{-5}	3.65×10^{-5}	3.6×10^{-5}	3.65×10^{-5}	3.65×10^{-5}	3.6×10^{-5}
Hexane	3.54×10^{0}	2.22×10^{-1}	3.28×10^{-3}	3.28×10^{-3}	3.28×10^{-3}	8.13×10^{-1}	3.28×10^{-3}	3.28×10^{-3}	8.13×10^{-1}
Manganese	2.84×10^{-1}	3.43×10^{-1}	—	—	—	1.51×10^{-2}	—	—	1.51×10^{-2}
Methyl Alcohol	6.62×10^{-1}	3.46×10^{-1}	6.84×10^{-2}	6.84×10^{-2}	6.84×10^{-2}	8.68×10^{-2}	6.84×10^{-2}	6.84×10^{-2}	8.68×10^{-2}
Methyl Ethyl Ketone	6.41×10^{0}	3.17×10^{0}	2.19×10^{-3}	2.19×10^{-3}	2.19×10^{-3}	3.47×10^{-2}	2.19×10^{-3}	2.19×10^{-3}	3.47×10^{-2}
Methyl Isobutyl Ketone	8.25×10^{0}	2.25×10^{0}	—	—	—	1.27×10^{-2}	—	—	1.27×10^{-2}
Methylene Chloride	1.53×10^{0}	1.19×10^{0}	—	—	—	8.23×10^{-1}	—	—	8.23×10^{-1}
Naphthalene	7.22×10^{-2}	3.08×10^{-2}	5.84×10^{-4}	5.84×10^{-4}	5.84×10^{-4}	6.08×10^{-4}	5.84×10^{-4}	5.84×10^{-4}	6.08×10^{-4}
Phenol	8.07×10^{-2}	1.37×10^{-2}	—	—	—	6.01×10^{-5}	—	—	6.01×10^{-5}
Phosphorus	2.97×10^{-3}	1.65×10^{-4}	—	—	—	1.6×10^{-6}	—	—	1.6×10^{-6}
Sodium Hydroxide	1.26×10^{-1}	1.26×10^{-1}	—	—	—	5.97×10^{-2}	—	—	5.97×10^{-2}
Toluene	3.91×10^{0}	7.66×10^{-1}	5.0×10^{-2}	5.0×10^{-2}	5.0×10^{-2}	9.2×10^{-1}	5.0×10^{-2}	5.0×10^{-2}	9.2×10^{-1}
Trichloroethylene	2.52×10^{1}	9.8×10^{0}	—	—	—	5.52×10^{-4}	—	—	5.52×10^{-4}
Vinyl Acetate	4.38×10^{-2}	5.9×10^{-3}	—	—	—	5.0×10^{-5}	—	—	5.0×10^{-5}
Xylene	1.46×10^{3}	1.22×10^{1}	1.58×10^{-1}	1.58×10^{-1}	1.58×10^{-1}	1.4×10^{0}	1.58×10^{-1}	1.58×10^{-1}	1.4×10^{0}

Table 5-6. (continued).

Pollutant	Baseline		Alternatives						
			Regionalization A			Regionalization B			
	Maximum Design Capacity	Actual[b]	4a	4b	4c	4d	4e	4f	4g
TOXIC POLLUTANTS (TONS PER YEAR)									
Nitric Acid	$1.1x10^{3}$	$2.6x10^{0}$	$5.1x10^{-2}$	$5.1x10^{-2}$	$1.2x10^{2}$	$6.5x10^{-2}$	$6.5x10^{-2}$	$1.5x10^{2}$	—
1,1,1-Trichloroethane	$8.0x10^{1}$	NA[c]	—	—	$7.0x10^{-1}$	—	—	$8.9x10^{-1}$	—
Benzene	$2.9x10^{1}$	$4.5x10^{0}$	—	—	$8.0x10^{-1}$	—	—	$1.0x10^{0}$	—
Ethanolamine	$2.2x10^{-2}$	$5.4x10^{-3}$	$1.5x10^{-3}$	$1.5x10^{-3}$	$1.5x10^{-3}$	$1.9x10^{-3}$	$1.9x10^{-3}$	$1.9x10^{-3}$	—
Ethyl Benzene	$2.6x10^{0}$	$1.1x10^{0}$	—	—	$8.0x10^{-4}$	—	—	$1.0x10^{-3}$	—
Ethylene Glycol	$6.8x10^{-1}$	$4.2x10^{-1}$	$2.3x10^{-2}$	$2.3x10^{-2}$	$4.3x10^{-2}$	$2.9x10^{-2}$	$2.9x10^{-2}$	$5.5x10^{-2}$	—
Formaldehyde	$4.6x10^{-2}$	$4.8x10^{-4}$	$3.6x10^{-6}$	$3.6x10^{-6}$	$3.6x10^{-5}$	$4.6x10^{-6}$	$4.6x10^{-6}$	$4.6x10^{-6}$	—
Glycol Ethers	$4.4x10^{-3}$	$2.0x10^{-4}$	$4.1x10^{-3}$	$4.1x10^{-3}$	$4.1x10^{-3}$	$5.2x10^{-3}$	$5.2x10^{-3}$	$5.2x10^{-3}$	—
Hexachloronapthalene	<0.01	NA[c]	$3.7x10^{-5}$	$3.7x10^{-5}$	$3.6x10^{-5}$	$4.7x10^{-5}$	$4.7x10^{-5}$	$4.6x10^{-5}$	—
Hexane	$3.5x10^{0}$	$2.2x10^{-1}$	$3.3x10^{-3}$	$3.3x10^{-3}$	$8.1x10^{-1}$	$4.2x10^{-3}$	$4.2x10^{-3}$	$1.0x10^{0}$	—
Manganese	$2.8x10^{-1}$	$3.4x10^{-1}$	—	—	$1.5x10^{-2}$	—	—	$1.9x10^{-2}$	—
Methyl Alcohol	$6.6x10^{-1}$	$3.5x10^{-1}$	$6.8x10^{-2}$	$6.8x10^{-2}$	$8.7x10^{-2}$	$8.6x10^{-2}$	$8.6x10^{-2}$	$1.1x10^{-1}$	—
Methyl Ethyl Ketone	$6.4x10^{0}$	$3.2x10^{0}$	$2.2x10^{-3}$	$2.2x10^{-3}$	$3.5x10^{-2}$	$2.8x10^{-3}$	$2.8x10^{-3}$	$4.4x10^{-2}$	—
Methyl Isobutyl Ketone	$8.3x10^{0}$	$2.3x10^{0}$	—	—	$1.3x10^{-2}$	—	—	$1.7x10^{-2}$	—
Methylene Chloride	$1.5x10^{0}$	$1.2x10^{0}$	—	—	$8.2x10^{-1}$	—	—	$1.0x10^{0}$	—
Naphthalene	$7.2x10^{-2}$	$3.1x10^{-2}$	$5.8x10^{-4}$	$5.8x10^{-4}$	$6.1x10^{-4}$	$7.4x10^{-4}$	$7.4x10^{-4}$	$7.7x10^{-4}$	—
Phenol	$8.1x10^{-2}$	$1.4x10^{-2}$	—	—	$6.0x10^{-5}$	—	—	$7.6x10^{-5}$	—
Phosphorus	$3.0x10^{-3}$	$1.7x10^{-4}$	—	—	$1.6x10^{-6}$	—	—	$2.0x10^{-6}$	—
Sodium Hydroxide	$1.3x10^{-1}$	$1.3x10^{-1}$	—	—	$6.0x10^{-2}$	—	—	$7.6x10^{-2}$	—
Toluene	$3.9x10^{0}$	$7.7x10^{-1}$	$5.0x10^{-2}$	$5.0x10^{-2}$	$9.2x10^{-1}$	$6.4x10^{-2}$	$6.4x10^{-2}$	$1.2x10^{0}$	—
Trichloroethylene	$2.5x10^{1}$	$9.8x10^{0}$	—	—	$5.5x10^{-4}$	—	—	$7.0x10^{-4}$	—
Vinyl Acetate	$4.4x10^{-2}$	$5.9x10^{-3}$	—	—	$5.0x10^{-5}$	—	—	$6.4x10^{-5}$	—
Xylene	$1.5x10^{3}$	$1.2x10^{1}$	$1.6x10^{-1}$	$1.6x10^{-1}$	$1.4x10^{0}$	$2.0x10^{-1}$	$2.0x10^{-1}$	$1.8x10^{0}$	—

Table 5-6. (continued).

			Alternatives			
			Centralization			
Pollutant	Maximum Design Capacity	Actual[b]	5a	5b	5c	5d
TOXIC POLLUTANTS (TONS PER YEAR)						
Nitric Acid	1.1×10^{3}	2.6×10^{0}	5.1×10^{-2}	5.1×10^{-2}	1.2×10^{2}	—
1,1,1-Trichloroethane	8.0×10^{1}	NA[c]	—	—	7.0×10^{-1}	—
Benzene	2.9×10^{1}	4.5×10^{0}	—	—	8.0×10^{-1}	—
Ethanolamine	2.2×10^{-2}	5.4×10^{-3}	1.5×10^{-3}	1.5×10^{-3}	1.5×10^{-3}	—
Ethyl Benzene	2.6×10^{0}	1.1×10^{0}	—	—	8.0×10^{-4}	—
Ethylene Glycol	6.8×10^{-1}	4.2×10^{-1}	2.3×10^{-2}	2.3×10^{-2}	4.3×10^{-2}	—
Formaldehyde	4.6×10^{-2}	4.8×10^{-4}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}	—
Glycol Ethers	4.4×10^{-3}	2.0×10^{-4}	4.1×10^{-3}	4.1×10^{-3}	4.1×10^{-3}	—
Hexachloronapthalene	<0.01	NA[c]	3.7×10^{-5}	3.7×10^{-5}	3.6×10^{-5}	—
Hexane	3.5×10^{0}	2.2×10^{-1}	3.3×10^{-3}	3.3×10^{-3}	8.1×10^{-1}	—
Manganese	2.8×10^{-1}	3.4×10^{-1}	—	—	1.5×10^{-2}	—
Methyl Alcohol	6.6×10^{-1}	3.5×10^{-1}	6.8×10^{-2}	6.8×10^{-2}	8.7×10^{-2}	—
Methyl Ethyl Ketone	6.4×10^{0}	3.2×10^{0}	2.2×10^{-3}	2.2×10^{-3}	3.5×10^{-2}	—
Methyl Isobutyl Ketone	8.3×10^{0}	2.3×10^{0}	—	—	1.3×10^{-2}	—
Methylene Chloride	1.5×10^{0}	1.2×10^{0}	—	—	8.2×10^{-1}	—
Naphthalene	7.2×10^{-2}	3.1×10^{-2}	5.8×10^{-4}	5.8×10^{-4}	6.1×10^{-4}	—
Phenol	8.1×10^{-2}	1.4×10^{-2}	—	—	6.0×10^{-5}	—
Phosphorus	3.0×10^{-3}	1.7×10^{-4}	—	—	1.6×10^{-6}	—
Sodium Hydroxide	1.3×10^{-1}	1.3×10^{-1}	—	—	6.0×10^{-2}	—
Toluene	3.9×10^{0}	7.7×10^{-1}	5.0×10^{-2}	5.0×10^{-2}	9.2×10^{-1}	—
Trichloroethylene	2.5×10^{1}	9.8×10^{0}	—	—	5.5×10^{-4}	—
Vinyl Acetate	4.4×10^{-2}	5.9×10^{-3}	—	—	5.0×10^{-5}	—
Xylene	1.5×10^{3}	1.2×10^{1}	1.6×10^{-1}	1.6×10^{-1}	1.4×10^{0}	—

a. Source: WSRC (1994a).

b. Maximum operational air pollutant emissions projected for baseline year 1995. Concentration estimates based on actual emissions from all SRS sources for calendar year 1990 plus maximum potential emissions for sources permitted through December 1992.

c. NA= Emissions data currently unavailable.

— No proposed incremental emissions.

consist of about 2×10^{-7} curies per year of cesium-137. Releases from dry storage activities under these alternatives would be somewhat less. For Alternative 5 where SRS would manage about 2,740 MTHM (3,020 tons) of spent fuel (versus about 206 to 257 MTHM [227 to 283 tons] for the other alternatives), the atmospheric releases of cesium-137 would be proportionally higher.

DOE used actual emissions from F- and H-Areas during 1985 and 1986, a period when the SRS was processing material through the separations facilities at close to maximum capacity to evaluate potential releases from spent nuclear fuel management activities. DOE believes that the isotopes released during this period, and their emission rates, represent maximum emissions that could occur under any of the alternatives (Table 5-7). The results of the analyses are presented in this section and the human health consequences are discussed in Section 5.12. Section 5.15 presents the analysis of the consequences of accidents.

Construction Emissions. Potential impacts to air quality from construction activities would include fugitive dust from the clearing of land, as well as exhaust emissions from support equipment (e.g., earth-moving vehicles, diesel generators). The amount of dust produced would be proportional to the land area disturbed for the new facilities, all of which would be located near the center of the Site. The areas affected by each alternative would be as follows:

- No Action - 0 acres
- Decentralization, 1992/1993 Planning Basis and Regionalization A (by fuel type) - 6 to 9 acres
- Regionalization B (by location) - 7 to 11 acres
- Centralization - 70 to 100 acres
- Shipping fuel offsite - 1 acre

DOE anticipates that overall construction impacts to air quality would be minimal and of a short duration (6 months to 3 years). The SRS sitewide compliance with state and Federal ambient air quality standards would not be affected by any construction-related activities associated with spent fuel management.

Table 5-7. Estimated maximum annual emissions (in curies) of radionuclides to the atmosphere from spent nuclear fuel management activities.

Radionuclide	Annual Emissions[a,b]
Tritium (elemental)	1.88×10^{5}[c]
Cesium-134	3.60×10^{-4}
Cesium-137	4.07×10^{-3}
Curium-244	2.00×10^{-4}
Cerium-141	1.83×10^{-3}
Cerium-144	3.11×10^{-2}
Americium-241	2.27×10^{-4}
Cobalt-60	4.00×10^{-6}
Plutonium-238	1.28×10^{-3}
Plutonium-239	4.01×10^{-4}
Strontium-90	1.39×10^{-2}
Rubidium-103	7.25×10^{-3}
Uranium-235	2.00×10^{-3}
Osmium-185	3.60×10^{-4}
Nibium-95	2.89×10^{-2}
Selenium-75	1.52×10^{-5}
Zirconium-95	1.68×10^{-2}
Rubidium-106	5.12×10^{-3}
Krypton-85	6.80×10^{5}
Carbon-14	2.80×10^{1}

a. Source: Hamby (1993).
b. Source terms are taken from 1985/86 F-/H-Area releases.
c. Historically, less than 10 percent of the atmospheric tritium releases have been from processing operations in the F-/H-Area Canyons.

5.7.1 Alternative 1 - No Action

The SRS would not process any spent nuclear fuel under the No Action alternative. Normal site baseline emissions would continue (Tables 5-3, 5-4, 5-5, 5-6 and 5-7). DOE would not construct any new facilities under this alternative.

5.7.2 Alternative 2 - Decentralization

Atmospheric emissions under two of the Decentralization options (dry storage and wet storage) would be similar to those for No Action. Those from the processing of the spent fuel (Option 2c) would be of somewhat higher concentrations (Tables 5-3, 5-4, 5-5, 5-6 and 5-7). The emissions would originate from existing facilities involved in the management of spent fuel under this alternative as well as new ones that DOE would construct (Figure 3-2).

5.7.3 Alternative 3 - 1992/1993 Planning Basis

Emissions to the atmosphere would be similar to those for Alternative 2 because the amount of fuel managed would be similar [223 and 220 MTHM (246 and 243 tons), Alternative 3 and Alternative 2 respectively] and the facilities required would be the same (Figure 3-2).

5.7.4 Alternative 4 - Regionalization

Regionalization A (by fuel type). Atmospheric emissions would be similar to the releases from Alternative 2 because of the similarity in volumes of fuel managed [213 and 220 MTHM (235 and 243 tons), respectively] and in the facilities involved (Figure 3-2).

Regionalization B (by location). Emissions would be somewhat higher than for Regionalization A for both dry and wet storage options if the SRS receives all the spent fuel in the eastern portion of the country, because the Site would manage about 20 percent more fuel. Atmospheric emissions from processing would not change from those under other alternatives because the amount of aluminum-clad fuel involved would be the same. Facility requirements would also be similar (Figure 3-2).

Shipping all of the current SRS inventory off the Site (Option 4g) would result in the lowest emissions to the atmosphere of any of the options under this alternative. These releases would result from the characterization and canning of the fuel prior to shipment.

5.7.5 Alternative 5 - Centralization

The atmospheric emissions resulting from centralizing all the spent nuclear fuel at the SRS would be the greatest of all the alternatives. The Site would manage about 2,740 MTHM (3,020 tons) of fuel. Releases from storage activities for centralization would be proportionally higher than for the other alternatives where the SRS would manage about 206 to 257 MTHM (227 to 283 tons) of spent fuel. However, emissions from processing under Alternative 5 would be similar to those under the other alternatives because the same amount of aluminum-clad fuel would be processed in each case. The facilities required under all three options would be similar in function (Figure 3-2) but of much larger capacity than for other alternatives.

Shipping all the SRS fuel to another site (Option 5d) would result in the lowest level of atmospheric releases of any alternative, similar to those under Regionalization B, Option 4g.

5.8 Water Quality and Related Consequences

SRS use of surface-water and groundwater resources under any of the alternatives would not substantially increase the volumes currently used for process, cooling, and domestic water on the Site. Table 5-8 summarizes the groundwater and surface water usage requirements for each alternative and option, and compares them to current SRS usages.

The Centralization Alternative (Option 5c), under which DOE would transfer all spent nuclear fuel to the SRS, would result in the largest amount of water use [approximately 378.5 million liters (100 million gallons) per year], which is a small amount compared to current SRS water requirements of approximately 89.7 billion liters (23.7 billion gallons) per year. This represents an increase of approximately 0.4 percent above current usage. Therefore, DOE anticipates that water use under any of the alternatives would have minimal impact on the water resources of the Site.

The impact on water quality of the operation of any of the alternatives would also be minimal. Existing SRS treatment facilities could accommodate all new spent fuel-related domestic and process wastewater streams. The expected total SRS flow volumes would still be well within the design capacities of the Site treatment systems. Because these plants would continue to meet National Pollutant Discharge Elimination System limits and reporting requirements, DOE expects no impact on the water quality of the receiving streams. The increased cooling water flows would also meet all discharge permit limits and would have minimal impacts on the receiving water.

Each of the alternatives would contribute to the very small releases of radionuclides that normal SRS operations discharge to the surface water through federally permitted wastewater outfalls.

Table 5-8. Annual groundwater and surface water usage requirements for each alternative.[a,b]

Alternative	Groundwater Usage per Year	Surface Water Usage per Year	Total Annual
Current SRS Usage	14.0 billion liters	75.7 billion liters	89.7 billion liters
No Action			
Option 1 - Wet Storage	35.1 million liters	None	35.1 million liters
Decentralization			
Option 2a - Dry Storage	48.7 million liters	6.1 million liters	54.8 million liters
Option 2b - Wet Storage	50.6 million liters	7.2 million liters	57.8 million liters
Option 2c - Processing[c]	48.7 million liters	310.8 million liters	359.5 million liters
Planning Basis			
Option 3a - Dry Storage	48.7 million liters	6.1 million liters	54.8 million liters
Option 3b - Wet Storage	50.6 million liters	7.2 million liters	57.8 million liters
Option 3c - Processing[c]	48.7 million liters	310.8 million liters	359.5 million liters
Regionalization - A			
Option 4a - Dry Storage	48.7 million liters	6.1 million liters	54.8 million liters
Option 4b - Wet Storage	50.6 million liters	7.2 million liters	57.8 million liters
Option 4c - Processing[c]	47.6 million liters	308.8 million liters	356.5 million liters
Regionalization - B			
Option 4d - Dry Storage	48.7 million liters	6.1 million liters	54.8 million liters
Option 4e - Wet Storage	50.6 million liters	7.2 million liters	57.8 million liters
Option 4f - Processing[c]	48.7 million liters	310.8 million liters	356.5 million liters
Option 4g - Ship Out[c]	38.1 million liters	3.0 million liters	41.1 million liters
Centralization			
Case 5a - Dry Storage	67.7 million liters	6.1 million liters	73.8 million liters
Case 5b - Wet Storage	69.6 million liters	7.2 million liters	76.8 million liters
Case 5c - Processing[c]	67.7 million liters	310.8 million liters	378.5 million liters
Case 5d - Ship Out[c]	38.1 million liters	3.0 million liters	41.1 million liters

a. Source: WSRC (1994b).
b. To convert liters to gallons, multiply by 0.26418.
c. First 10 years only.

Table 5-9 summarizes the estimated maximum amounts of radioactivity that could be released to the Savannah River in liquid effluents from normal spent nuclear fuel management activities. DOE used actual liquid releases from F- and H-Area during 1985 and 1986 to estimate potential releases that could occur during spent fuel management activities. DOE believes the isotopes and amounts released during this period are representative of releases that could occur during processing under any of the alternatives. This is because 1985 and 1986 represent periods when the F- and H-Area separations facilities operated at or near peak capacity to process spent nuclear fuel. Estimated releases from wet or dry storage would be less than these amounts. Consequently, the estimated releases given in Table 5-9 represent the upper limit of liquid radiological releases that DOE expects as a result of spent

Table 5-9. Estimated maximum liquid radiological releases (in curies) to the Savannah River from spent nuclear fuel management activities.

Radionuclide	Annual Release[a,b]
Tritium	$1.3 \times 10^{4,c}$
Strontium-90	2.4×10^{-1}
Iodine-129	2.2×10^{-2}
Cesium-137	1.1×10^{-1}
Plutonium-239	7.0×10^{-3}

a. Source: Hamby (1993).
b. Source terms are taken from 1985/86 F-/H-Area releases.
c. Less than 1 percent of this quantity was from processing operations in F-/H-Area.

nuclear fuel management activities. The consequences to human health due to these releases are discussed in Section 5.12, Occupational and Public Health and Safety.

Construction of new facilities under any alternative would require amounts of water that would be only a very small percentage of the current daily water use at the SRS. Good engineering practice measures would prevent sediment runoff or spills of fuel or chemicals. Therefore, construction activities should have no impact on surface or groundwater quality at the Site.

DOE also analyzed the potential impacts of accidents in F- and H-Areas on surface and groundwater quality. The analysis evaluated two types of accidental releases: one to the ground surface (e.g., overflow of a wet storage pool) and another directly to the subsurface (e.g., failure of a pool liner). Because pool water could contain some radionuclides, but would not contain any toxic or harmful chemicals, the following evaluation addresses only the consequences of radionuclide releases.

A release of pool water onto the ground from the Receiving Basin for Offsite Fuels, in H-Area, would not flow directly into any stream or other surface-water body. The building is in a graded, gravel-covered area among other buildings and alongside a railroad spur and access road. A tank farm surrounded by an earthen berm is immediately to the south. A channelized drainage ditch begins approximately 244 meters (800 feet) west of the basin building and passes through culverts under a railroad line and Road E before emptying into a tributary of Fourmile Branch about 500 meters (1,650 feet) from the Receiving Basin. The grading at the Site would contain a small volume of water overflowing the basin in the immediate area of the building. In the unlikely event that a larger spill reached the drainage ditch to the west, DOE could contain the water by blocking either of the two culverts through which the drainage ditch passes. After containing the spilled water, DOE could remove and properly dispose of it. DOE would design and construct new facilities containing storage pools in a manner that would confine any overflow or other surface release of pool water. Therefore, DOE believes that there will be no direct release to surface water from spills of pool water at an existing or potential facility.

An overflow from a pool could reach the groundwater by slowly flowing downward from the surface through the unsaturated zone until it reached the water table, which is 9 to 15 meters (30 to 50 feet) below the grade in the F- and H-Areas. Overflow water would take several years to reach the water table, based on a vertical velocity of between 0.9 and 2.1 meters (3 to 7 feet) per year (DOE 1987). As discussed in the following paragraphs, once in the groundwater, a plume would take many years to reach either of the closest surface-water bodies, Fourmile Branch to the south or Upper Three Runs Creek to the north.

DOE has calculated the travel times of groundwater in the F- and H-Areas based on specific information on the hydraulic conductivity, the hydraulic gradient, and the effective porosity of aquifers in this area (WSRC 1993a) and on the use of Darcy's Law. Water would take between 16 and 500 years to travel 1.6 kilometers (1 mile) toward Fourmile Branch or Upper Three Runs Creek. These estimates of travel time agree with values obtained from the results of DOE modeling studies performed on the F- and H-Areas (Geotrans 1993; appended to WSRC 1993a). The reason for this wide range of potential travel time is that the hydraulic conductivity of aquifer materials is highly variable and can vary in the same aquifer by several orders of magnitude. This slow movement through the subsurface, either vertically through the unsaturated zone or horizontally within the aquifer, would facilitate the removal of radionuclides from the spill plume through a number of processes. These include radioactive decay, trapping of particulates in the soil, and ion exchange and adsorption by the soil (Hem 1989). DOE believes that travel time of a contaminant plume through the subsurface in the F- or H-Area or in the adjacent representative host site would be such that no radionuclides would reach Fourmile Branch, Upper Three Runs Creek, or any other surface-water body by this route. For the same reasons, no radioactive contaminants introduced into the subsurface in these areas would move off the Site in groundwater.

DOE does not believe that releases of radionuclides such as those described above would reach SRS drinking-water sources that lie in deep aquifers under the Site. These aquifers are several hundred feet below the ground surface, and a number of thick aquifers and aquitards separate them from the water table aquifer (see Section 4.8). In addition to the distances and the presence of confining layers, vertical flow in the intervening stratified sedimentary aquifers is slow in comparison to horizontal flow. Radionuclide contamination of offsite drinking water sources is even more unlikely given the depth of their source aquifers, the distances involved, and the attenuation of contaminants in the soils, as described above.

DOE also evaluated a second kind of unintentional release in the F- or H-Area, a direct leak to the subsurface from a breach in a storage pool during routine operations. The analysis assumed a 19-liter (5-gallon)-per-day leak as a result of secondary containment or piping failure at a new state-of-the-art wet storage and fuel transfer facility (Creed 1994). The analysis assumed further that the leak would go undetected for 1 month, a conservative assumption given the sensitivity of the leak detection equipment that these new facilities would require. The reliability and sensitivity of the leak detection devices would be equal to or superior to those required by the U.S. Nuclear Regulatory Commission (NRC 1975) for spent nuclear fuel storage facilities in commercial nuclear power plants. DOE would require spent nuclear fuel storage pools (whether fuel unloading pools or storage basins) to have leak detection monitoring devices, pool water level monitors, and radiation monitors designed to alarm both locally and in a continuously staffed central location. Constant process monitoring, mass balance, and facility design (including double-walled containment of vessels and piping) would also be used by DOE to limit operational releases from new wet storage facilities, including fuel unloading pools and storage basins, to near zero.

To provide a common basis for analysis of spent nuclear fuel alternatives at its various sites, DOE developed a generic infrastructure design for a hypothetical spent nuclear fuel complex (Hale 1994). This design includes proposed criteria for temporary wet storage basins, fuel loading and unloading pools, and transfer canals.

Based on the design criteria in Hale (1994), a leak from one of these basins if constructed in F- or H-Area could result in the introduction of radionuclide-contaminated water into the ground at depths as much as 13.4 meters (44 feet) below grade. Such a release would go directly to the water table aquifer or to the unsaturated zone above it, depending on the depth of the water table. In either case, the processes governing the slow plume movement (i.e., the hydraulic conductivity, hydraulic gradient, and effective porosity of aquifers in the F- and H-Areas) and the processes resulting in the attenuation of contaminants and radionuclides (i.e., radioactive decay, trapping of particulates in the soil, ion exchange in the soil, and adsorption to soil particles) described in the previous paragraphs would also prevent or mitigate impacts to surface-or groundwater resources from releases of this type. There could be localized contamination of groundwater in the surface aquifer in the immediate vicinity of the storage facilities. This aquifer is not used as a source of drinking water. DOE believes that no radionuclide contamination of deeper confined aquifers that are sources of onsite or offsite drinking water could occur from a release of this type. And, as noted earlier, these wet storage facilities would be equipped with state-of-the-art leak detection devices, pool level monitors, and radiation monitors that would limit and mitigate any subsurface releases.

5.8.1 Alternative 1 - No Action

5.8.1.1 Option 1 - Wet Storage. During operations under this alternative, current levels of water usage would not change. Nor would changes occur in thermal discharges from cooling water or the quantity or quality of radioactive and nonradioactive wastewater effluents.

The viable accidents under this alternative would be a release of pool water onto the ground surface or a breach of the liner of the wet storage basins in which the spent nuclear fuel would be stored. As discussed above, radionuclides in the released water would enter the water table aquifer but would not reach any surface-water or any drinking water aquifer on or off the SRS. Basin water contains no toxic or hazardous chemicals. Therefore, accidental releases from the basins would have minimal impacts on surface- and groundwater resources.

Spills of chemicals would not reach surface- or groundwater due to existing proper engineering design and environmental controls, and to rapid containment and cleanup.

5.8.2 Alternative 2 - Decentralization

Operations under either the dry or wet storage option for the Decentralization alternative would increase Site water usage by less than 0.1 percent above current levels. Processing would increase use by about 0.4 percent. Release of nonradioactive and radioactive materials to surface waters would increase only slightly and would be well within discharge permit limits and DOE dose limits. There would be no releases to groundwater during normal operations. Overall impacts to water quantity and water quality would be minimal.

Impacts to water resources due to accidental releases onto the ground or into the subsurface would also be minimal as explained above. Potential contamination would be limited to the surface aquifer.

5.8.3 Alternative 3 - 1992/1993 Planning Basis

DOE expects that the impacts to water resources under the dry storage, wet storage, and processing cases for this alternative would be similar to those described for the same options under Alternative 2, Decentralization. Overall impacts would be minimal.

5.8.4 Alternative 4 - Regionalization

DOE expects that the impacts to water resources under the three options for regionalization by fuel type (Regionalization A) would be similar to those described for the same options under Alternative 2, Decentralization. Regionalization B (by geographic location) would result in impacts somewhat greater than those for Alternative 2 because the SRS would have to manage an additional 37 MTHM (41 tons) of spent fuel. In either case, overall impacts would be minimal. For Option 4g, shipping all SRS fuel to Oak Ridge Reservation, impacts to water resources would be the smallest of any alternative, similar to those for Option 5d - Centralization.

5.8.5 Alternative 5 - Centralization

The first three options for this alternative - dry storage (Option 5a), wet storage (Option 5b), and processing (Option 5c) - assume that DOE would transfer all spent nuclear fuel to the SRS for management. The impacts of operations to water resources under these options would be similar in nature to the impacts for the same options under Alternative 2, Decentralization, as described in Section 5.8.2. However, the extent of the impacts would be greater because the number and size of facilities that DOE would construct and operate and the quantities of fuel it would manage would be larger than those for any other alternative. Even so, DOE expects the overall impacts of construction and operation to be minor. For example, the total volume of water that the SRS would withdraw for construction, cooling, processing, and domestic use under any of these three options would not exceed approximately 378.5 million liters (100 million gallons) per year. This requirement would be approximately 0.4 percent of the 89.7 billion liters (23.7 billion gallons) that the SRS currently uses annually.

Similarly, DOE believes that the overall impacts of accidents under any of these three options would be minor, even though the number and size of the facilities would be greater under this alternative than for any other. Radionuclides released during an accident would not affect any surface-water or any drinking water aquifer. However, surface aquifer resources would receive contamination in the area of any release.

For Option 5d (shipping the spent nuclear fuel off the Site), impacts to water resources would be smaller than those for any other alternative or option. DOE would have to build only one new facility (for fuel characterization) and the spent fuel would remain at SRS only for the first part of the 40-year management period. Overall impacts would be minimal.

5.9 Ecology

DOE expects that construction impacts, which would include loss of some wildlife habitat due to land clearing, would be greatest under the Centralization Alternative, Dry Storage option. Representative impacts from operations would include disturbance and displacement of animals caused by movement and noise of personnel, equipment, and vehicles; however, these impacts would be minor under all the proposed alternatives. Construction and operation would not disturb any critical or sensitive habitat, nor would they affect any wetland areas. Releases of radionuclides to the environment from any of the proposed alternatives would be small and would not be expected to accumulate in aquatic or terrestrial ecosystems or measurably affect the health or viability of plant and animal communities.

5.9.1 Alternative 1 - No Action

Under this alternative, DOE could refurbish or modify existing wet storage facilities and would confine any activity to these facilities. As a consequence, DOE expects no impacts to ecological resources. Impacts of operations under this alternative would be minimal, limited to some minor disturbance of animals by vehicular traffic.

5.9.2 Alternative 2 - Decentralization

5.9.2.1 Option 2a - Dry Storage. This option would require some new construction, but any construction activity would occur either within the boundaries of F- and H-Areas, which are already heavily developed, or adjacent to them. As a result, this construction would have little or no impact on ecological resources. There would be no impacts to wetlands, threatened or endangered species, socially or commercially important species (such as the eastern wild turkey), or disturbance-sensitive species (such as wood warblers and vireos). Impacts of operations under this option would be limited to some minor disturbance of animals by slight increases in vehicular traffic. No threatened, endangered, or candidate species occur in the area of operations. Species likely to be disturbed or killed by vehicles (e.g., cotton rat, gray squirrel, opossum, and white-tailed deer) are common to ubiquitous in the area. Overall impact to ecological resources would be minimal.

5.9.2.2 Option 2b - Wet Storage. Construction impacts would be similar to those described for dry storage (Option 2a). Impacts of operations under this option would also be similar to those described for dry storage (Option 2a). Overall impacts to ecological resources would be minimal.

5.9.2.3 Option 2c - Processing and Storage. Construction and operations impacts for this option would also be similar to those for dry storage (Option 2a). Overall impacts would still be minimal.

5.9.3 Alternative 3 - 1992/1993 Planning Basis

Both construction and operational impacts for the three options under this alternative would be similar to those described for Alternative 2 - Decentralization. Overall impacts would be minimal.

5.9.4 Alternative 4 - Regionalization

Under the Regionalization A alternative, impacts to ecological resources would be minimal as described for Alternative 2. Impacts due to the Regionalization B options would be somewhat greater due to the larger volume of spent fuel that the SRS would manage. Overall impacts would still be minimal, however.

The smallest impacts would occur under Option 4g because DOE would ship all spent fuel off the Site.

5.9.5 Alternative 5 - Centralization

5.9.5.1 Option 5a - Dry Storage. The discussion that follows assumes that any facility development would take place in an area that does not contain any pristine wetlands, old growth timber, threatened and endangered species, or designated critical habitat. More specifically, because the upland areas south and east of H-Area are dominated by planted pine (primarily loblolly and slash) stands, the discussion of impacts assumes that any facility development in support of spent nuclear fuel management would take place in an area of 5- to 40-year-old pines. Finally, the analysis assumes that any facility development would require a site-specific National Environmental Policy Act (NEPA) review as required under 10 CFR Part 1021 and in accordance with the Council on Environmental Quality's NEPA implementing regulations (CFR 1991b).

The proposed interim dry storage facility and support facilities, requiring approximately 0.28 square kilometers (70 acres) to 0.4 square kilometer (100 acres) of land, would be built somewhere within the largely wooded roughly 2.8 square kilometer (700-acre) area south and east of H-Area west of F-Road, and north of Fourmile Branch. This area has a number of advantages; among them: it would be relatively easy to connect with existing utilities (gas, water, sewer); it would minimize the amount of supporting infrastructure (e.g., railroad spurs, access roads, and transmission lines) that would have to be built; and it would enable DOE to consolidate spent nuclear fuel management activities in an area that has been altered many times over the years by farming (before 1951) and timber management activities (after 1951).

Construction activities would result in the clearing of as much as approximately 0.4 square kilometer (100 acres) of planted 5- to 40-year-old loblolly or slash pine for new facilities on the undeveloped representative host site south and east of H-Area. This land clearing would involve a relatively small number of loggers and heavy equipment operators, but probably would drive most birds and larger, more mobile animals from the area. Some smaller, less mobile animals, such as turtles, toads, lizards, mice, and voles, probably would be killed. Aside from the loss of 0.28 to 0.4 square kilometer (70-100 acres) of planted pines that provide habitat for a limited number of reptiles, birds, and mammals, construction impacts would be minor.

Any land clearing and timber harvesting conducted on the undeveloped host site would be carefully planned and conducted according to widely accepted Best Management Practices to minimize erosion and soil loss and to prevent impacts to downgradient wetlands and streams. DOE and SRS policy is to achieve "no net loss" of wetlands. DOE has issued a guidance document, *Information for Mitigation of Wetlands Impacts at the Savannah River Site* (DOE 1992), for project planners that puts forth a practical approach to wetlands protection that begins with avoidance of impacts (if possible), moves to minimization of impacts (if avoidance is impossible), and requires compensatory measures (wetlands restoration, creation, enhancement, or acquisition) in the event that impacts cannot be avoided.

In the event that new facility development was required, DOE would perform predevelopment surveys to ensure that its activities would not affect threatened and endangered species or sensitive habitats. To the extent practicable, land clearing and timber harvesting would be restricted to times of the year when songbirds and game birds were not nesting or rearing young. In South Carolina, most songbirds nest, rear, and fledge young from March to September (Sprunt and Chamberlain 1970). Quail, dove, and wild turkey in the region normally nest and fledge young during the spring and summer (Sprunt and Chamberlain 1970).

No threatened or endangered plants or animals are known to be present in the area under consideration for development. Construction activities probably would not affect two small wetlands (Carolina bays) lying in the east-central portion of the undeveloped host site. Construction activities would not affect plant and animal diversity locally or regionally, because the managed loblolly and slash pine stands that would be removed are not unique, nor do they provide habitat for any protected, sensitive, unusual, or Federally listed plant or animal species.

Impacts of operations under this option would be similar to, but slightly greater than, those described for Option 2a. Overall impacts to ecological resources would be minor.

5.9.5.2 Option 5b - Wet Storage. Construction impacts under this option would be less than those described for Option 5a because less land area would be required for new facilities. Impacts of operations under this case would be similar to those described for Option 5a. Overall impacts to ecological resources would be minor.

5.9.5.3 Option 5c - Processing and Storage. Construction impacts under this case would be similar to those described for Option 5a. This case would require the largest number of workers of all the cases under consideration. It would result in more noise, more traffic, and a generally higher level of disturbance to terrestrial wildlife (specifically reptiles, songbirds, and small and large mammals) accustomed to feeding, foraging, perching, hunting, nesting, or denning in the area. Some animals would be driven from the area permanently, while others probably would become accustomed to the increased noise and activity levels, and would return to the area. Overall impacts to ecological resources would be minor.

5.9.5.4 Option 5d - Shipment off the Site. Construction impacts under this case would be smaller than those for any other alternative, excluding Alternative 1 - No Action. Impacts of operation under this case would also be minimal, limited to some minor disturbances of animals by vehicular traffic. Overall impacts to ecological resources would be minimal.

5.10 Noise

As described in Section 4.10, noises generated on the SRS do not travel off the Site at levels that affect the general population. Therefore, SRS noise impacts for each alternative would be limited to noise resulting from the transportation of personnel and materials to and from the Site that could affect nearby communities and from onsite sources that could affect some wildlife near these sources. DOE would address the effects of noise on wildlife near spent nuclear fuel management facilities under any alternative in a project-specific NEPA evaluation.

Transportation noises would be a function of the size of the workforce (i.e., an increased workforce would produce increased employee traffic and corresponding increases in deliveries by truck and rail and a decreased workforce would produce decreased employee traffic and corresponding decreases in deliveries). The analysis of traffic noise took into account railroad noise and noise from the major roadways that provide access to the SRS. DOE does not expect the number of freight trains per day in the region and through the Site to change as a result of any of the alternatives, although some trains could be dedicated to the transport of spent nuclear fuel. Rail shipments of spent nuclear fuel, regardless of the alternative, would not substantially increase the rail traffic on the CSX line through the SRS. Therefore, vehicles used to transport employees and personnel on roadways would be the principal sources of community noise impacts. This analysis used the day-night average sound level (DNL) to assess community noise, as suggested by the Environmental Protection Agency (EPA 1974; 1982) and the Federal Interagency Committee on Noise (FICON 1992). The analysis based its estimate of the change in day-night average sound level from the baseline noise level for each alternative on the projected changes in employment and traffic levels. The baseline levels are those for 1995. The analysis also considered the combination of construction and operation employment. The traffic noise analysis considered SC 125 and SC 19, both of which are used to access the SRS. Changes in noise level below 3 decibels would not be likely to result in a change in community reaction (FICON 1992).

DOE projects no new employment due to operations for any of the alternatives. Some additional construction jobs may be required but overall SRS employment would not exceed the 1995 baseline levels, except for Alternatives 5a, 5b, and 5c. The maximum Site employment of about 20,000 jobs would occur in 1995 for all alternatives except 5a, 5b, and 5c for which the peak would occur in about 2002 due to a peak in construction employment. The general decrease in employment after 1995 could result in some decrease in vehicle trips to and from the Site. There would be at most a few truck trips per day to and from the Site carrying spent nuclear fuel under any of the alternatives. This increase in truck trips would not result in a perceptible increase in traffic noise levels along the routes to the SRS. The day-night average sound level along SC 125 and SC 19 and other access routes would probably decrease slightly except in the peak construction years under Alternatives 5a, 5b, and 5c, as a result of the overall decrease in employment levels at the SRS after 1995. DOE expects no change in the community reaction to noise along these routes. Consequently, no mitigation efforts are necessary.

5.11 Traffic and Transportation

This section discusses the consequences of both the onsite transportation of spent nuclear fuel and the increased traffic patterns due to construction activities at the SRS. Traffic due to operations of spent nuclear fuel facilities will remain at or below current Site levels because workers for the new activities will be drawn from the existing SRS workforce. The consequences of the transportation of spent fuel between the SRS and other DOE sites are described in Appendix I of Volume 1 of this Environmental Impact Statement (EIS).

5.11.1 Traffic

Traffic impacts would be bound by Alternative 5b (Centralization - Wet Storage) which would result in the greatest number of additional construction workers (and vehicles) onsite. Level of service, a measure of traffic flow, was estimated for each road to and from the SRS. Traffic delays could be experienced at SC 19 and SC 230 intersections during peak hours. However, the number of construction vehicles in support of spent nuclear fuel construction activities would contribute less than 17 percent (HNUS 1994) to the total traffic flow. Therefore, the change in level of service due to Alternative 5b would be minimal.

5.11.2 Transportation

This section discusses the potential radiological consequences due to incident free transportation and accidents during transport. All SRS onsite shipments are carried out by rail.

5.11.2.1 Onsite Spent Nuclear Fuel Shipments. DOE based the number of fuel shipments on the amount and type of spent nuclear fuel stored at various SRS locations and the final storage location or disposition specified in the spent nuclear fuel alternatives. The number of shipments from each location was determined by dividing the amount of spent nuclear fuel at each location by the capacity of the shipping cask. Individual shipments from the various facilities were summed to obtain the total number of shipments for each alternative (HNUS 1994).

Onsite shipments are those that originate and terminate at the SRS. Movements of spent nuclear fuel within functional areas (e.g., H-Area or F-Area) are operational transfers, not onsite shipments; therefore, this analysis does not consider them.

5.11.2.2 Incident-Free Transportation Analysis. Under each alternative, DOE analyzed incident-free (normal transport) radiological impacts to transport vehicle crews and members of the general public from onsite rail shipments. The analysis calculated occupational radiation doses to the transport vehicle crew members (four locomotive operators). Because the general public does not have immediate access to areas where the SRS would transport spent nuclear fuel, the analysis assumed that any general public dose is to escorted individuals on the Site waiting at any of several train crossings at the time a fuel shipment passed. The analysis calculated radiological doses to the general public using the RISKIND (Yuan et al. 1993) computer code. The results are presented in Table 5-10.

The magnitude of incident-free consequence depends on the dose rate on the external surface of the transport vehicle, the exposure time, and the number of people exposed. For each receptor, the analysis assumed the external dose rate 2 meters (6.6 feet) from the shipping cask was 100 millirem per hour (HNUS 1994), which is the SRS procedurally-allowed maximum dose rate during onsite fuel shipments. Actual receptor dose rates would depend on receptor distance from the shipping cask [5 meters (16.4 feet) for the general public]. The duration of exposure would depend on the transport vehicle speed and the number of shipments. In addition, occupational exposure time would depend on the distance of each shipment.

The analysis calculated health effects measured as the number of latent cancer fatalities (LCFs) by multiplying the resultant occupational and general public doses by risk factors of 4×10^{-4} and 5×10^{-4} latent cancer fatalities per person-rem (DOE 1993a), respectively.

Table 5-10 summarizes the collective doses (person-rem) and health effects (latent cancer fatalities) associated with the incident-free onsite shipment of spent nuclear fuel at the SRS. Collective

Table 5-10. Collective doses and health effects for onsite, incident-free spent nuclear fuel shipments by alternative.

Option	Occupational (person-rem)	General Public (person-rem)	Number of LCFs[a] Occupational	General Public
No Action				
Option 1b -Wet Storage	$1.5x10^{0}$	$1.4x10^{-1}$	$6.0x10^{-4}$	$7.0x10^{-5}$
Decentralization				
Option 2a - Dry Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 2b - Wet Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 2c - Processing	$5.3x10^{-1}$	$3.7x10^{-2}$	$2.1x10^{-4}$	$1.9x10^{-5}$
Planning Basis				
Option 3a - Dry Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 3b - Wet Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 3c - Processing	$5.3x10^{-1}$	$3.7x10^{-2}$	$2.1x10^{-4}$	$1.9x10^{-5}$
Regionalization				
Option 4a - Dry Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 4b - Wet Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 4c - Processing	$5.3x10^{-1}$	$3.7x10^{-2}$	$2.1x10^{-4}$	$1.9x10^{-5}$
Option 4d - Dry Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 4e - Wet Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 4f - Processing	$5.3x10^{-1}$	$3.7x10^{-2}$	$2.1x10^{-4}$	$1.9x10^{-5}$
Option 4g - Ship Out	NA[b]	NA[b]	NA[b]	NA[b]
Centralization				
Option 5a - Dry Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 5b - Wet Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 5c - Processing	$5.3x10^{-1}$	$3.7x10^{-2}$	$2.1x10^{-4}$	$1.9x10^{-5}$
Option 5d - Ship Out	NA[b]	NA[b]	NA[b]	NA[b]

a. LCF = latent cancer fatality.
b. NA = not applicable.

doses and latent cancer fatalities for members of the public would be approximately a factor of 10 less than those for the occupational worker. The data indicate that the lowest collective doses and lowest latent cancer fatality would be associated with the Processing option under the Decentralization, Planning Basis, Regionalization, and Centralization alternatives.

5.11.2.3 Transportation Accident Analysis. DOE analyzed radiological impacts from potential accidents to both the onsite maximally exposed individual (MEI), and offsite members of the general public from onsite rail shipments. The analysis calculated doses using the RISKIND (Yuan et al. 1993) computer code with site-specific meteorology, demographics, and spent fuel activity. Risk was calculated using site-specific rail accident rates and accident probabilities (HNUS 1994).

The magnitude of accident consequence would depend on the amount of radioactive material to which the individual(s) was exposed, the exposure time, and the number of people exposed. The analysis assumed that the maximum reasonably foreseeable amount of radioactive material for the type of spent fuel shipped on the SRS was released (HNUS 1994). The assumed duration of exposure for each receptor was 2 hours. The assumed maximally exposed individual was an SRS worker downwind of the accident at distances of 50 and 100 meters (164 and 330 feet).

The analysis calculated offsite exposure using both rural and suburban population density-specific census data. The rural and suburban population densities have an average of 6 persons per square kilometer and 244 persons per square kilometer, respectively. The west-northwest sector has the highest population density within 80 kilometers (50 miles) of the SRS.

The analysis used site-specific meteorology at the 50th and 95th percentile to determine dose consequences. Joint probability includes both the event frequency and the probability of the maximum reasonably foreseeable type of accident occurring.

The analysis calculated health effects measured as the number of latent cancer fatalities by multiplying the resultant occupational and general public doses by the risk factors of 4×10^{-4} and 5×10^{-4} latent cancer fatalities per person-rem (DOE 1993a), respectively. Risk was calculated by multiplying the resultant doses by the joint probability of 1×10^{-4} (HNUS 1994).

Tables 5-11 and 5-12 summarize the collective doses and associated latent cancer fatalities for postulated onsite rail accidents with subsequent releases of radioactive material to the environment. The dose consequences of an accidental release of radioactive material was assessed for the 95th and typical 50th percentile meteorological conditions (i.e., those that would result in lower doses 95 and 50 percent of the time, respectively). In all cases the estimated number of latent cancer fatalities would be low.

5.11.3 Onsite Mitigation and Preventative Measures

All onsite shipments must be in compliance with DOE Savannah River Directive Implementation Instruction 5480.3, "Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes." DOE, DOE-SR, or the Nuclear Regulatory Commission (NRC) must approve packages used for onsite shipments with a certificate of

Table 5-11. Impacts on maximally exposed individual from spent nuclear fuel transportation accident on the Savannah River Site.

Dose Percentile	Distance (meters)	Dose to MEI[a] (rem)	Number of LCFs[b] per year	Risk
50 percent	100	0.16	6.4×10^{-5}	1.6×10^{-5}
95 percent	50	0.37	1.5×10^{-4}	3.7×10^{-5}

a. MEI = maximally exposed individual.
b. LCF = latent cancer fatality.

Table 5-12. Impacts on offsite population from spent nuclear fuel transportation accident on the Savannah River Site.

Population Density Category	Dose Percentile	Offsite Population Dose (person-rem)	Number of LCFs[a] per year	Risk
Rural	50th	1.7	8.7×10^{-4}	1.7×10^{-4}
Rural	95th	7.1	3.6×10^{-3}	3.6×10^{-3}
Suburban	50th	5.2	2.6×10^{-3}	2.6×10^{-3}
Suburban	95th	21.3	1.1×10^{-2}	1.1×10^{-2}

a. LCF = latent cancer fatality.

compliance. If DOE or NRC has not certified an onsite package as Type B, the shipper must establish administrative controls and site-mitigating circumstances that will ensure package integrity. The administrative and emergency response considerations must provide sufficient control so that accidents would not result in loss of containment, shielding, or criticality; or the uncontrolled release of radioactive material would not create a hazard to the health and safety of the public or workers.

In the event of an accident, SRS has established an emergency management program. This program incorporates activities associated with emergency planning, preparedness, and response.

5.12 Occupational and Public Health and Safety

5.12.1 Radiological Health

This human health effects analysis relied principally on data on F- and H-Area emissions documented for the 1985, 1986, and 1993 operating years (Marter 1986; 1987; WSRC 1994d). During the 1985-1986 period, F- and H-Areas processing facilities operated at high capacity; DOE believes, therefore, that these emissions represent conservative estimates as to the emissions that could result from spent nuclear fuel management activities at the SRS. This air and surface-water emissions information defined the source terms for the baseline evaluation (No Action alternative) of health effects discussed in this section. To estimate health effects, this analysis defined six human receptor groups:

- The F- and H-Area workers assigned to F- and H-Area operations involving nuclear materials
- The F- and H-Area workers assigned to the Receiving Basin for Offsite Fuels for storage operations
- The maximally exposed individual residing at the SRS boundary
- The projected 1994 offsite population of 628,200 persons residing within an 80-kilometer (50-mile) radius of F- and H-Areas
- The maximally exposed individual potentially affected by SRS surface-water emissions
- The approximate offsite population of 65,000 persons whom SRS surface-water emissions could affect.

With the exception of the worker group, this analysis calculated exposures for the remaining four receptor groups using the baseline source terms as input data to automated atmospheric and surface-water transport, human intake, and human dosimetry models configured for routine use at SRS (Hamby 1994). The analysis estimated worker exposures using averaged dosimetry data recorded for F- and H-Area workers from 1983 through 1987 and Receiving Basin for Offsite Fuels workers for 1993 (Matheny 1994), corrected for an assumed occupancy factor of 0.25 (i.e., a worker could be potentially exposed during one-quarter of his/her shift). This correction was applied to the 1983-1987 data only. At the SRS, the waterborne exposure pathway does not exist for the worker receptor group because Site drinking water is drawn from deep aquifers unaffected by any radiological releases.

The analysis developed incremental receptor group exposure estimates (millirem per year, person-rem per year; effective dose equivalent) based on spent fuel quantities for each of the nonbaseline alternatives (i.e., Alternatives 2 through 5) and their options by applying calculated ratios of metric tons of heavy metal (MTHM) for each alternative and option compared to the No Action alternative. DOE used these ratios as incremental scaling factors to estimate exposures under each option. The calculation of the MTHM ratios used the data presented in Table 3-1. Table 5-13 lists the results of the exposure estimate calculations. Since these incremental exposures include contributions to the effective dose equivalent from existing (No Action) spent fuel management at the SRS, the change in health effects for each alternative can be estimated as the difference between the alternatives presented.

The analysis calculated the potential health effects expressed in the exposed receptor groups consistent with risk determination guidance issued by the DOE Office of NEPA Oversight (DOE 1993a) and International Commission on Radiological Protection Publication 60 (ICRP 1991). For exposed individuals and populations, the potential health effect (detriment) of interest is latent fatal cancer. For exposed individuals, this analysis presents the health effect as the maximum incremental probability for detriment expression; for exposed populations, it presents the annual incremental detriment incidence. For completeness, it also provides the "project life" (i.e., 40 years) detriment incidence as the annual incidence multiplied by 40. Table 5-14 (worker) and Table 5-15 (maximally exposed individual and offsite population) summarize the health effects calculations.

The Centers for Disease Control and Prevention is conducting a comprehensive reconstruction of historic offsite doses associated with SRS operations. The results of this investigation are not yet available.

5.12.2 Nonradiological Health

DOE used the operations air quality data listed in Tables 5-3, 5-4, 5-5 and 5-6 (and Table 8 of WSRC 1994a) to evaluate health impacts associated with potential exposure to the following two compound classes: criteria pollutants and toxic pollutants. The analysis evaluated two hypothetical receptor locations: (1) a worker in S-Area and (2) a maximally exposed individual at the SRS boundary. However, it was unnecessary to postulate an intake of criteria pollutant or toxic compounds by these receptors because airborne concentration standards are available for these compounds.

Tables 5-3 and 5-4 list 8 criteria pollutants and 23 toxic compounds. The toxic compounds were classified as carcinogens and noncarcinogens consistent with Environmental Protection Agency carcinogenicity group (weight of evidence) designations published in the Integrated Risk Information

Table 5-13. Incremental radioactive contaminant annual exposure summary.

Alternative	Onsite Workers[a] (mrem/ year)[c]	Onsite Workers[a] (person-rem/ year)	MEI Offsite[a,b,d] (mrem/year) Air	MEI Offsite[a,b,d] (mrem/year) Water	Offsite Population[a,d] (person-rem/ year) Air	Offsite Population[a,d] (person-rem/ year) Water
No Action - Wet Storage (Option 1)	100	0.2	$9x10^{-8}$	$3x10^{-8}$	$4x10^{-6}$	$6x10^{-7}$
Decentralization - Dry Storage (Option 2a)	83	0.2	$8x10^{-8}$	$2x10^{-8}$	$3x10^{-6}$	$5x10^{-7}$
Decentralization - Wet Storage (Option 2b)	104	0.2	$9x10^{-8}$	$3x10^{-8}$	$4x10^{-6}$	$6x10^{-7}$
Decentralization - Processing (Option 2c)	145	70	0.4	0.1	14	2.2
Planning Basis - Dry Storage (Option 3a)	84	0.2	$8x10^{-8}$	$2x10^{-8}$	$3x10^{-6}$	$5x10^{-7}$
Planning Basis - Wet Storage (Option 3b)	105	0.2	$1x10^{-7}$	$3x10^{-8}$	$4x10^{-6}$	$6x10^{-7}$
Planning Basis - Processing (Option 3c)	147	71	0.4	0.1	15	2.2
Regionalization A - Dry Storage (Option 4a)	83	0.2	$8x10^{-8}$	$2x10^{-8}$	$3x10^{-6}$	$5x10^{-7}$
Regionalization A - Wet Storage (Option 4b)	103	0.2	$9x10^{-8}$	$3x10^{-8}$	$4x10^{-6}$	$6x10^{-7}$
Regionalization A - Processing (Option 4c)	148	76	0.4	0.1	16	2.4
Regionalization B - Dry Storage (Option 4d)	105	0.2	$1x10^{-7}$	$3x10^{-8}$	$4x10^{-6}$	$6x10^{-7}$
Regionalization B - Wet Storage (Option 4e)	131	0.3	$1x10^{-7}$	$4x10^{-8}$	$5x10^{-6}$	$7x10^{-7}$
Regionalization B - Processing (Option 4f)	175	74	0.4	0.1	15	2.3
Regionalization B - Ship Out (Option 4g)	<100	<0.2	$<9x10^{-8}$	$<3x10^{-8}$	$<4x10^{-6}$	$<6x10^{-7}$
Centralization - Dry Storage (Option 5a)	1,102	2.2	$1x10^{-6}$	$3x10^{-7}$	$4x10^{-5}$	$6x10^{-6}$
Centralization - Wet Storage (Option 5b)	1,377	2.8	$1x10^{-6}$	$4x10^{-7}$	$5x10^{-5}$	$8x10^{-6}$
Centralization - Processing (Option 5c)	1,422	79	0.4	0.1	16	2.4
Centralization - Ship Out (Option 5d)	<100	<0.2	$<9x10^{-8}$	$<3x10^{-8}$	$<4x10^{-6}$	$<6x10^{-7}$

a. Insignificant digits are displayed for comparison purposes only.
b. MEI = maximally exposed individual.
c. The DOE administrative dose limit is 2,000 mrem (DOE 1994a).
d. Data is provided separately for the air and water exposure pathways because the receptors are not co-located.

Table 5-14. Incremental fatal cancer incidence and maximum probability for workers.

Alternative	Annual Incidence[a]	40-Year Incidence	Maximum Probability
No Action - Wet Storage (Option 1)	$8x10^{-5}$	$3x10^{-3}$	$4x10^{-5}$
Decentralization - Dry Storage (Option 2a)	$7x10^{-5}$	$3x10^{-3}$	$3x10^{-5}$
Decentralization - Wet Storage (Option 2b)	$8x10^{-5}$	$3x10^{-3}$	$4x10^{-5}$
Decentralization - Processing (Option 2c)	$3x10^{-2}$	1	$6x10^{-5}$
Planning Basis - Dry Storage (Option 3a)	$7x10^{-5}$	$3x10^{-3}$	$3x10^{-5}$
Planning Basis - Wet Storage (Option 3b)	$8x10^{-5}$	$3x10^{-3}$	$4x10^{-5}$
Planning Basis - Processing (Option 3c)	$3x10^{-2}$	1	$6x10^{-5}$
Regionalization A - Dry Storage (Option 4a)	$7x10^{-5}$	$3x10^{-3}$	$3x10^{-5}$
Regionalization A - Wet Storage (Option 4b)	$8x10^{-5}$	$3x10^{-3}$	$4x10^{-5}$
Regionalization A - Processing (Option 4c)	$3x10^{-2}$	1	$6x10^{-5}$
Regionalization B - Dry Storage (Option 4d)	$8x10^{-5}$	$3x10^{-3}$	$4x10^{-5}$
Regionalization B - Wet Storage (Option 4e)	$1x10^{-4}$	$4x10^{-3}$	$5x10^{-5}$
Regionalization B - Processing (Option 4f)	$3x10^{-2}$	1	$7x10^{-5}$
Regionalization B - Ship Out (Option 4g)	$<8x10^{-5}$	$<3x10^{-3}$	$<4x10^{-5}$
Centralization - Dry Storage (Option 5a)	$9x10^{-4}$	$4x10^{-2}$	$4x10^{-4}$
Centralization - Wet Storage (Option 5b)	$1x10^{-3}$	$4x10^{-2}$	$5x10^{-4}$
Centralization - Processing (Option 5c)	$3x10^{-2}$	1	$6x10^{-4}$
Centralization - Ship Out (Option 5d)	$<8x10^{-5}$	$<3x10^{-3}$	$<4x10^{-5}$

a. Number of latent fatal cancers over a lifetime which could be attributed to one year of spent nuclear fuel management activities.

System (IRIS) data base (DOE 1994b). For purposes of health effects analysis, carcinogens are those compounds designated Group A (human carcinogens), Group B1 (probable human carcinogen, limited evidence in human studies), Group B2 (probable human carcinogen, inadequate evidence or no data from human studies), and Group C (possible human carcinogen). Using this designation, three of the 23 toxic compounds are carcinogens: benzene (Group A), formaldehyde (Group B1), and methylene chloride (Group B2).

Carcinogen health effects are expressed as the incremental probability of an individual developing cancer, assuming a lifetime (70 years) of exposure to the carcinogen. DOE used cancer risk (slope) factors published in IRIS (Integrated Risk Information System) to obtain unit risk factors (risk per concentration) needed to calculate incremental probability. Carcinogens with insufficient (i.e., incomplete or unavailable carcinogen assessment data) information listed in the Integrated Risk Information System data base precluded a quantitative risk assessment; this analysis evaluated them as noncarcinogens.

Table 5-15. Incremental fatal cancer incidence and maximum probability for the maximally exposed individual and offsite population (air and water pathways).

Alternative	Population Annual Incidence[a]	Population 40-Year Incidence	MEI Maximum Probability
No Action - Wet Storage (Option 1)			
Air	$2x10^{-9}$	$7x10^{-8}$	$4x10^{-14}$
Water	$3x10^{-10}$	$1x10^{-8}$	$1x10^{-14}$
Decentralization - Dry Storage (Option 2a)			
Air	$2x10^{-9}$	$6x10^{-8}$	$4x10^{-14}$
Water	$2x10^{-10}$	$9x10^{9}$	$1x10^{-14}$
Decentralization - Wet Storage (Option 2b)			
Air	$2x10^{9}$	$8x10^{-8}$	$5x10^{-14}$
Water	$3x10^{-10}$	$1x10^{-8}$	$2x10^{-14}$
Decentralization - Processing (Option 2c)			
Air	$7x10^{-3}$	0.3	$2x10^{-7}$
Water	$1x10^{3}$	$4x10^{-2}$	$6x10^{-8}$
Planning Basis - Dry Storage (Option 3a)			
Air	$2x10^{-9}$	$6x10^{-8}$	$4x10^{-14}$
Water	$2x10^{-10}$	$9x10^{-9}$	$1x10^{-14}$
Planning Basis - Wet Storage (Option 3b)			
Air	$2x10^{-9}$	$8x10^{-8}$	$5x10^{-14}$
Water	$3x10^{-10}$	$1x10^{-8}$	$2x10^{-14}$
Planning Basis - Processing (Option 3c)			
Air	$7x10^{-3}$	0.3	$2x10^{-7}$
Water	$1x10^{-3}$	$4x10^{-2}$	$6x10^{-8}$
Regionalization A - Dry Storage (Option 4a)			
Air	$2x10^{-9}$	$6x10^{-8}$	$4x10^{-14}$
Water	$2x10^{-10}$	$9x10^{-9}$	$1x 10^{-14}$
Regionalization A - Wet Storage (Option 4b)			
Air	$2x10^{-9}$	$8x10^{-8}$	$5x10^{-14}$
Water	$3x10^{-10}$	$1x10^{-8}$	$2x10^{-14}$
Regionalization A - Processing (Option 4c)			
Air	$8x10^{-3}$	0.3	$2x10^{-7}$
Water	$1x10^{-3}$	$5x10^{-2}$	$6x10^{-8}$
Regionalization B - Dry Storage (Option 4d)			
Air	$2x10^{-9}$	$8x10^{-8}$	$5x10^{-14}$
Water	$3x10^{-10}$	$1x10^{-8}$	$2x10^{-14}$
Regionalization B - Wet Storage (Option 4e)			
Air	$2x10^{-9}$	$1x10^{-7}$	$6x10^{-14}$
Water	$4x10^{-10}$	$1x10^{-8}$	$2x10^{-14}$
Regionalization B - Processing (Option 4f)			
Air	$8x10^{-3}$	0.3	$2x10^{-7}$
Water	$1x10^{-3}$	$5x10^{-2}$	$6x10^{-8}$
Regionalization B - Ship Out (Option 4g)			
Air	$<2x10^{-9}$	$<7x10^{-8}$	$<4x10^{-14}$
Water	$<3x10^{-10}$	$<1x10^{-8}$	$<1x10^{-14}$

Table 5-15. (continued).

Alternative	Population Annual Incidence[a]	Population 40-Year Incidence	MEI Maximum Probability
Centralization - Dry Storage (Option 5a)			
Air	$2x10^{-8}$	$8x10^{-7}$	$5x10^{-13}$
Water	$3x10^{-9}$	$1x10^{-7}$	$2x10^{-13}$
Centralization - Wet Storage (Option 5b)			
Air	$3x10^{-8}$	$1x10^{-6}$	$6x10^{-13}$
Water	$4x10^{-9}$	$2x10^{-7}$	$2x10^{-13}$
Centralization - Processing (Option 5c)			
Air	$8x10^{-3}$	0.3	$2x10^{-7}$
Water	$1x10^{-3}$	$5x10^{-2}$	$6x10^{-8}$
Centralization - Ship Out (Option 5d)			
Air	$<2x10^{-9}$	$<7x10^{-8}$	$<4x10^{-14}$
Water	$<3x10^{-10}$	$<1x10^{-8}$	$<1x10^{-14}$

a. Number of latent fatal cancers over a lifetime that could be attributed to one year of spent nuclear fuel management activities.

This analysis evaluated noncarcinogenic and priority pollutant compound health effects by adding hazard quotients to obtain a hazard index. The hazard quotient is the ratio of compound concentration or dose to a Reference Concentration (RfC) or Dose (RfD) (EPA 1989). The regulatory standard used in this analysis was the more stringent of the following: (1) Occupational Safety and Health Administration (OSHA) 8-hour permissible exposure limit (PEL), (2) American Conference of Governmental Industrial Hygienists (ACGIH) threshold limit value (TLV), or (3) State of South Carolina air quality standards. The use of the noncancer hazard index assumed a level of exposure (i.e., RfC) below which adverse health effects are unlikely. The hazard index is not a statistical probability; therefore it cannot be interpreted as such.

Table 5-16 summarizes nonradiological health effects attributable to atmospheric emissions of toxic and criteria pollutant compounds. Because no hazard index value would exceed unity (1.0), adverse health effects are unlikely under any alternative.

5.12.3 Industrial Safety

This section describes the following measures of impact for workplace hazards: (1) total reportable injuries and illnesses and (2) fatalities in the work force. This analysis considers injury/illness and fatality incidence rates for construction workers separately because of the relatively

Table 5-16. Nonradiological annual incremental health effects summary.

Alternative	Worker Cancer Probability[a]	Worker Hazard Index	MEI Cancer Probability[a,b]	MEI Hazard Index
No Action - Wet Storage (Option 1)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Decentralization - Dry Storage (Option 2a)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Decentralization - Wet Storage (Option 2b)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Decentralization - Processing (Option 2c)	Insufficient data	$6x10^{-3}$	Insufficient data	$5x10^{-4}$
Planning Basis - Dry Storage (Option 3a)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Planning Basis - Wet Storage (Option 3b)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Planning Basis - Processing (Option 3c)	Insufficient data	$6x10^{-3}$	Insufficient data	$5x10^{-4}$
Regionalization A - Dry Storage (Option 4a)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Regionalization A - Wet Storage (Option 4b)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Regionalization A - Processing (Option 4c)	Insufficient data	$6x10^{-3}$	Insufficient data	$5x10^{-4}$
Regionalization B - Dry Storage (Option 4d)	Insufficient data	$2x10^{-6}$	Insufficient data	$3x10^{-7}$
Regionalization B - Wet Storage (Option 4e)	Insufficient data	$2x10^{-6}$	Insufficient data	$3x10^{-7}$
Regionalization B - Processing (Option 4f)	Insufficient data	$8x10^{-3}$	Insufficient data	$6x10^{-4}$
Regionalization B - Ship Out (Option 4g)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Centralization - Dry Storage (Option 5a)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Centralization - Wet Storage (Option 5b)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Centralization - Processing (Option 5c)	Insufficient data	$6x10^{-3}$	Insufficient data	$5x10^{-4}$
Centralization - Ship Out (Option 5d)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$

a. Insufficient data exists in the IRIS data base to perform a quantitative inhalation cancer risk assessment.
b. MEI = maximally exposed individual.

more hazardous nature of construction work. Table 5-17 lists the incidence of injuries/illnesses and fatalities for construction and non-construction workers. These data are for the highest employment year (i.e., maximum hours worked in any year from 1994 through 2035, assuming 2,000 hours per worker) (WSRC 1994b). This analysis used the average occupational injury/illness and fatality incidence rates experienced by DOE and its contractors from 1988 through 1992 to calculate the incidence of industrial hazards listed in Table 5-17 (DOE 1993b).

Table 5-17. Incremental industrial hazard maximum annual incidence summary.

Alternative	Construction Injuries and Illnesses	Construction Fatalities	Nonconstruction Injuries and Illnesses	Nonconstruction Fatalities
No Action - Wet Storage (Option 1)	92	<1	159	<1
Decentralization - Dry Storage (Option 2a)	71	<1	159	<1
Decentralization - Wet Storage (Option 2b)	71	<1	159	<1
Decentralization - Processing (Option 2c)	66	<1	159	<1
Planning Basis - Dry Storage (Option 3a)	71	<1	159	<1
Planning Basis - Wet Storage (Option 3b)	82	<1	159	<1
Planning Basis - Processing (Option 3c)	66	<1	159	<1
Regionalization A - Dry Storage (Option 4a)	82	<1	159	<1
Regionalization A - Wet Storage (Option 4b)	82	<1	159	<1
Regionalization A - Processing (Option 4c)	66	<1	159	<1
Regionalization B - Dry Storage (Option 4d)	89	<1	199	<1
Regionalization B - Wet Storage (Option 4e)	102	<1	199	<1
Regionalization B - Processing (Option 4f)	82	<1	199	<1
Regionalization B - Ship Out (Option 4g)	22	<1	159	<1
Centralization - Dry Storage (Option 5a)	316	1	159	<1
Centralization - Wet Storage (Option 5b)	337	1	159	<1
Centralization - Processing (Option 5c)	316	1	159	<1
Centralization - Ship Out (Option 5d)	22	<1	159	<1

5.13 Utilities and Energy

The existing capacities and distribution systems at the SRS for electricity, steam, water, and domestic wastewater treatment are adequate to support any of the five alternatives. Table 5-18 summarizes estimates of the annual requirements for electricity, steam, and domestic wastewater treatment for each alternative and case, and compares them to current SRS usage of these resources. Table 5-8 lists information on water usage by alternative. The utility and energy requirements for all

Table 5-18. Estimates of annual electricity, steam, and domestic wastewater treatment requirements for each alternative.[a,b]

Alternative	Electricity Usage (megawatt hours per year)	Steam Usage (kilograms per year)[c]	Domestic Wastewater Treatment (liters per year)[d]
Current SRS Usage	659,000	1.7 billion	690 million
1. No Action			
Option 1 - Wet Storage	1,400	11.3 million	35.1 million
2. Decentralization			
Option 2a - Dry Storage	19,400	16.7 million	48.7 million
Option 2b - Wet Storage	22,400	14.4 million	50.6 million
Option 2c - Processing	56,400	19.1 million	48.7 million
3. 1992/1993 Planning Basis			
Option 3a - Dry Storage	19,400	16.7 million	48.7 million
Option 3b - Wet Storage	22,400	14.4 million	50.6 million
Option 3c - Processing	56,400	19.1 million	48.7 million
4. Regionalization - A			
Option 4a - Dry Storage	24,400	16.7 million	48.7 million
Option 4b - Wet Storage	27,400	14.4 million	50.6 million
Option 4c - Processing	67,400	16.5 million	47.6 million
Regionalization - B			
Option 4d - Dry Storage	24,400	16.7 million	48.7 million
Option 4e - Wet Storage	27,400	14.4 million	50.6 million
Option 4f - Processing	56,400	19.1 million	48.7 million
Option 4g - Ship Out	11,400	11.7 million	38.1 million
5. Centralization			
Option 5a - Dry Storage	44,400	16.7 million	67.7 million
Option 5b - Wet Storage	47,400	14.4 million	69.6 million
Option 5c - Processing	110,400	19.1 million	67.7 million
Option 5d - Ship Out	11,400	11.7 million	38.1 million

a. Source: WSRC (1994b).
b. Water requirements are shown in Table 5-8.
c. To convert kilograms to pounds, multiply by 2.2046.
d. To convert liters to gallons, multiply by 0.26418.

the alternatives represent a small percentage of current requirements. No new generation or treatment facilities would be necessary; connections to existing networks would require only short tie-in lines. Increases in SRS fuel consumption would be minimal because overall activity on the Site would not increase due to changes in the SRS mission and the general reduction in employment levels. The overall impacts of any of the alternatives on the SRS utilities and energy resources would be minimal.

The smallest increase in demand would result from the No Action alternative, which would be similar to current spent nuclear fuel-related requirements at the SRS. The largest increases would be due to the centralization of spent nuclear fuel at the SRS (Alternative 5). Alternative 5 would result in a maximum additional electrical demand of about 110,400 megawatt-hours annually (Option 5c), and an increased steam consumption of about 19.1 million kilograms (42.1 million pounds) per year (Option 5c). Water requirements would also be greatest under this Alternative (Table 5-8). Annual withdrawals of Savannah River water for cooling purposes would reach about 310.8 million liters (82.1 million gallons) and groundwater usage for domestic and processing purposes would total approximately 69.6 million liters (18.4 million gallons). The volume of domestic wastewater requiring treatment would range from approximately 35 to 70 million liters (9 to 18 million gallons) per year. This additional water usage amounts to an increase of about 10 percent over current SRS water requirements.

Among the three management options, processing would result in the greatest increase in demand on utilities and energy in comparison to either the dry or wet storage options. In general, dry and wet storage would be similar in their requirements of these resources.

5.14 Materials and Waste Management

This section discusses potential impacts of the management of materials and wastes associated with the implementation of alternatives identified for spent nuclear fuel management. Sections 5.7 and 5.12 (Air Quality and Occupational and Public Health and Safety, respectively) discuss the impacts of hazardous and toxic materials as they relate to routine operations and accidents.

DOE has projected rates and volumes of waste and impacts of waste generation at SRS for low-level, transuranic, and high-level wastes for each of the alternatives for spent nuclear fuel management. Table 5-19 summarizes the estimated annual average and total volume of these three waste types that each alternative would produce during a 40-year management period. The discussion

Table 5-19. Annual average and total volume (cubic meters)[d] of radioactive wastes produced under each alternative during the 40-year interim management period.[a]

Alternative	Low-level waste[b]		Transuranic waste		High-level waste[c]	
	Average	Total	Average	Total	Average	Total
1. No Action						
Option 1 - Wet Storage	400	16,000	17	700	0.4	4
2. Decentralization						
Option 2a - Dry Storage	400	16,000	18	720	0.4	4
Option 2b - Wet Storage	400	16,000	18	720	0.4	4
Option 2c - Processing	800	32,000	19	760	2.3	23
3. 1992/1993 Planning Basis						
Option 3a - Dry Storage	400	16,000	18	720	0.4	4
Option 3b - Wet Storage	400	16,000	18	720	0.4	4
Option 3c - Processing	750	30,000	19	760	1.7	17
4. Regionalization - A						
Option 4a - Dry Storage	400	16,000	17	700	0.4	4
Option 4b - Wet Storage	400	16,000	17	700	0.4	4
Option 4c - Processing	790	31,600	18	720	2.3	23
4. Regionalization - B						
Option 4d - Dry Storage	400	16,000	17	700	0.4	4
Option 4e - Wet Storage	400	16,000	17	700	0.4	4
Option 4f - Processing	790	31,600	18	720	2.3	23
Option 4g - Ship Out	400	4,000	18	180	0	0
5. Centralization						
Option 5a - Dry Storage	400	16,000	16	640	0	0
Option 5b - Wet Storage	400	16,000	20	800	2.3	23
Option 5c - Processing	800	32,000	20	800	2.3	23
Option 5d - Ship Out	400	4,000	18	180	0	0

a. Based on WSRC (1994b).
b. Source: WSRC (1994c).
c. Figures are for the initial 10-year period when most processing would be completed.
d. To convert cubic meters to cubic yards multiply by 1.307.

below also identifies the impacts that the waste produced by spent nuclear fuel activities would have on the existing SRS capacity to manage each waste type.

DOE has not developed estimates of low-level mixed, hazardous, or solid sanitary wastes that spent nuclear fuel management activities at the SRS could generate, although it is anticipated that these activities would produce these waste types only in limited quantities. Further, the discussions in Section 5.14.2 related to the impacts of spent fuel management wastes on the SRS waste capacities do not include considerations of wastes that will result from Site cleanup because assessments for these activities are still underway and will undergo NEPA review as part of the SRS Waste Management Environmental Impact Statement (DOE 1995).

Volume 1 of this spent nuclear fuel EIS provides information concerning the major Federal environmental laws and regulations, Executive Orders, and DOE Orders that apply to pollution prevention at the Savannah River Site. The DOE views source reduction as the first priority in its pollution prevention program, followed by an increased emphasis on recycling. Source reduction will reduce the waste management burden while eliminating the potential for future liability and cleanup. Recycling and using recycled materials will conserve resources and landfill space. Waste treatment and disposal are considered only when prevention or recycling is not possible or practical. Since creating a Savannah River Site waste minimization program (the precursor of the SRS pollution prevention program) in 1990, the amounts of wastes of all types (excluding low-level wastes, which are a by-product of environmental restoration activities) generated have decreased, with greatest reductions in hazardous and mixed wastes (Hoganson and Miles 1994).

5.14.1 Alternative Comparison

The first four alternatives would generate similar amounts of radioactive waste because the activities that produce the wastes would be similar under each of the alternatives. Most of the low-level and transuranic wastes would be generated during the first part of the 40-year management period while DOE was transferring existing inventory and renovating the Receiving Basin for Offsite Fuels and a reactor basin. The characterization and canning of the current inventory prior to placement into storage would also result in some waste generation. Once in storage, management activities would produce only small amounts of radioactive waste for the rest of the 40-year period.

The dry- and wet-storage options would both produce about 16,000 cubic meters (20,912 cubic yards) of low-level waste and between 640 cubic meters (836 cubic yards) and 800 cubic meters (1,046 cubic yards) of transuranic waste during the 40-year management period. Both options would generate small amounts of high-level waste. The processing of the existing aluminum-clad fuels and storage of the others (the third option under each alternative) would generate all three types of waste: low-level and high-level wastes in appreciably greater volumes, and transuranic waste in slightly-greater volumes.

Alternative 5 (excluding the Ship Out option) could result in somewhat larger volumes of radioactive waste than the other four alternatives. However, any increase in waste would not be directly proportional to the larger amounts of fuel that would be managed on the Site, because most of the originating sites would characterize and can their fuel prior to shipment so that it could be placed directly into storage at the SRS. Therefore, the radioactive wastes produced during centralization at the Site would come from the initial fuel transfer and pool renovations and from characterizing and canning small amounts of new fuel. The processing of existing aluminum-clad fuels would produce the same types and volumes of waste as for the other alternatives.

The option for shipping the SRS inventory off the Site for regionalization or centralization elsewhere would also result in the production of some radioactive waste. This would occur during characterization and canning prior to shipment and would generate the smallest volumes of waste of any alternative action: 4,000 cubic meters (5,228 cubic yards) of low-level waste and 180 cubic meters (235 cubic yards) of transuranic waste. This waste would be produced only during the initial 10 years of the management period.

5.14.2 Impact on the SRS Waste Management Capacity

The impact of spent nuclear fuel activities on SRS waste management capacities would be minimal because the Site could accommodate the waste with existing and planned radioactive waste storage and disposal facilities. DOE would transfer high-level waste to the F/H Tank Farms for volume reduction and then to the Defense Waste Processing Facility (DWPF) for conversion into a borosilicate glass form suitable for prolonged storage. The SRS would use the Consolidated Incineration Facility, once operational, to treat the low-level waste. This facility has sufficient permitted capacity [105,500 cubic meters (137,889 cubic yards) per year] to treat the anticipated volume of these materials. However, actual through-put volume is dependent upon operational variables and waste characteristics. The F/H Effluent Treatment Facility would treat liquid low-level waste. This facility has sufficient design process capacity [598 million liters (158 million gallons) per year] to treat the anticipated volumes of these materials. DOE would manage the transuranic wastes with existing and planned storage capacity.

5.15 Accident Analysis

Operations involving the receipt, handling, processing, or storing of spent nuclear fuel would involve radioactive materials or toxic chemicals. These materials would be received, treated, stored, transferred between facilities, disposed of on the Site, and shipped off the Site. Under certain circumstances, these materials could be involved in an accident.

An accident is a series of unexpected or undesirable events initiated by equipment failure, human error, or a natural phenomenon such as severe weather, earthquake, or volcanism. These events can cause the release of either radioactive or chemically toxic materials inside a facility or to the environment.

This section summarizes analyses of possible accidents involving spent nuclear fuel operations at the SRS. To provide a perspective on potential accidents, this section summarizes various accidents associated with spent nuclear fuel activities that have occurred at the SRS (historic accidents) and reviews previous accident analyses for Site operations. This section uses the results of previous analyses as a baseline for determining the impacts for the alternatives that involve new facilities. For each alternative, this section discusses the accidents with the largest point estimates of risk (radiological impacts in terms of potential fatal cancers x frequency of the initiating event).

The facilities considered for each alternative are either existing facilities for which the approved safety analyses were used, or new facilities (WSRC 1994b) for which existing safety analysis results were substituted by evaluating the type of accident(s) that could be postulated to occur based on the projected function of the facility. Two facilities that contain very small amounts of contact-handled spent nuclear fuel, Buildings 331-M and 773-A, were not included in this analysis because accidents analyzed for the major facilities would bound the consequences of possible accidents in these two locations.

This section addresses historic accidents, facility radiological accidents, chemical hazard accidents, and secondary impacts. Section 5.11 addresses onsite transportation accidents.

5.15.1 Historic Accidents at the Savannah River Site

Impacts from accidents can involve fatalities, injuries, or illness. Fatalities can be prompt (immediate) such as in construction accidents or latent (delayed) such as an increase in latent fatal cancers due to radiation exposure. Section 5.12 addresses worker injuries, illnesses, and the potential for increased cancer risk anticipated from normal operations of the facilities. Nonradiation accidents have dominated impacts to workers at the SRS (Durant et al. 1987); impacts to the public from historic SRS accidents have been negligible.

The SRS has maintained an operational event data base on its facilities since the 1950s. This data base currently contains approximately 450,000 entries including data on the Receiving Basin for

Offsite Fuel, the principal wet storage pool facility at the SRS; and both F-and H-Area Canyons. For this EIS, DOE reviewed the data base to identify historic spent nuclear fuel-related accidents at these facilities. Fuel cutting events, fuel handling events, and various liquid releases related to spent nuclear fuel management over the 40-year operating history of the SRS were examined. The purpose of the data base review was to provide an historic perspective on the types of accidents that have occurred at the SRS. Events representative of fuel failures include higher than expected contamination levels in fuel storage basin water and evidence of fuel canister cracking at a weld. Fuel handling incidents were due in large part to crane operator errors or crane and handling equipment failures. The data base also includes reports of incorrect fuel cropping, where the active region of fuel was exposed under water. These historical events provided a basis for the selection of representative accidents covering the spectrum of spent nuclear fuel management activities. No significant offsite impacts have resulted from these historic occurrences.

5.15.2 Potential Facility Accidents

The SRS spent nuclear fuel alternatives have the potential for radiological accidents (see Attachment A, Table A-2) that could affect the health and safety of workers and the public. The concerns and characteristics that are common to these accidents would be common regardless of whether the cause were a natural phenomenon or human error. For health effects to occur, an accident must allow a release of hazardous material to, or an increase in radiation levels in, the facility or the environment. The released material must be transported to locations frequented by humans. The quantities of hazardous materials that reach locations where people are and the ways they interact with people are important factors in the determination of health effects.

A number of studies have investigated the ways in which radioactivity reaches humans, how the body absorbs and retains it, and the resulting health effects. The International Commission on Radiological Protection has made specific recommendations for estimating these health effects (ICRP 1991). This organization is the recognized body for establishing standards for the protection of workers and the public from the effects of radiation exposure. Health effects include acute damage (up to and including death) and latent effects, including cancers and genetic damage. An SRS-developed computer code, AXAIR89Q, estimates potential radiation doses to maximally exposed individuals or population groups from accidental releases of radionuclides.

The AXAIR89Q code is a highly automated site-specific environmental dispersion and dosimetry code for postulated airborne releases. The environmental dispersion models used are based on NRC

Regulatory Guide 1.145 (NRC 1983). The exposure pathways considered in the AXAIR89Q code include inhalation of radionuclides and gamma irradiation from the radioactive plume.

Doses from the inhalation of radionuclides in air depend on the amount of radionuclides released; the dispersion factor; the physical, chemical, and radiological characteristics of the radionuclides; and various biological parameters such as breathing rate and biological half-life. The AXAIR89Q code uses a conservative breathing rate of 12,000 cubic meters (424,000 cubic feet) per year for adults. The dose commitment factors used in the environmental dosimetry code, as described in the following section, are from *Internal Dose Conversion Factors for Calculation of Dose to the Public* (DOE 1988).

External gamma radiation doses from the traveling plume depend on the spatial distribution of the radionuclides in the air, the energy of the radiation, and the extent of shielding. The AXAIR89Q code takes no credit for shielding in calculating doses. The code calculates gamma doses using a nonuniform Gaussian model, which has more realistic modeling than doses from the conventional uniform semi-infinite plume model.

In addition to using the worst sector, 99.5 percentile meteorology, conservative breathing rates, and taking no credit for shielding, the AXAIR89Q code also takes no credit for the probable plume rise from stack releases. Therefore, the offsite maximum individual doses calculated by AXAIR89Q provide conservative bounding estimates of radiological consequences to exposed individuals and populations from postulated accidental atmospheric releases.

AXAIR89Q has been validated for compliance to accepted standards for such software. Attachment A, Accident Analysis, discusses AXAIR89Q and its predecessor, AXAIR. When used in conjunction with models for predicting health effects, the results from AXAIR89Q can be compared with other site-specific codes such as RSAC-5, because both codes provide relative radionuclide concentrations based on the guidance provided in NRC Regulatory Guide 1.145.

This section summarizes the potential for radiological accidents and their consequences for the cases under each alternative. Attachment A describes the methodology and assumptions used in the assessment; describes radiological accident scenarios in more detail; provides source terms and references used to estimate the doses and impacts for each alternative and case; and includes scaling factors that the DOE decisionmaker can apply to the source term or dose for each facility associated with a case.

DOE assessed the potential impacts from a selected spectrum of radiological release accidents, ranging from low (1×10^{-6} event per year) to high (more than 1 event per year) frequencies of occurrence, along with the associated impacts (doses and potential latent fatal cancers) that could result. The accidents used as references are attributed to individual facilities based on their functions and processes (see Attachment A, Table A-3), not to specific cases or alternatives. This enables a comparison of alternatives depending on which facilities support a specific case or alternative. Figure 5-1 is a flowchart for the preparation of accident analysis information. No new analyses occurred because existing documentation adequately supports a quantitative or qualitative estimation of potential impacts, as required by the National Environmental Policy Act of 1969. The assessment of postulated radiological accidents associated with spent nuclear fuel at the SRS indicates that the highest point estimate of risk to the public within 80 kilometers (50 miles) of the Site would be 1.4×10^{-3} latent fatal cancer per year. The estimated dose to the same population from all causes, including natural background sources, would be about 19,000 person-rem per year (DOE 1990), which could cause about nine latent fatal cancers per year in the same population. For perspective, natural background radiation sources would result in approximately 6,000 times the risk associated with the largest consequence accident postulated in this EIS for the various spent nuclear fuel management alternatives.

DOE did not quantitatively analyze the potential health effects for SRS workers less than 100 meters (328 feet) from radiological accidents. Computer codes used to calculate radiological doses can experience potentially large errors as a source disperses throughout a building. However, DOE did carry out a qualitative evaluation of the potential radiological effects to SRS workers in the immediate vicinity of an accident related to spent fuel management. DOE estimates that the consequences of an accident for the most part would result in higher than normal radiation doses. However, no fatalities would occur except in the event of an inadvertent criticality in FB-Line, where up to four fatalities may result. This evaluation is discussed in more detail in Section A.2.6.2 of Attachment A.

5.15.2.1 Alternative 1 - No Action. This alternative identifies the minimum actions deemed necessary for continued safe and secure management of spent nuclear fuel at the SRS. As explained in Chapter 3, this is not a *status quo* condition. Spent nuclear fuel would be maintained close to defueling or current storage locations with minimal facility upgrade or equipment replacement. Only local transport would occur. SRS activities required to safely store spent nuclear fuel would continue. This alternative would require SRS to place corroded and pitted fuel elements in cans to minimize spread of material into the pool. DOE estimated potential radiological accident impacts that could occur under this alternative using existing DOE-approved safety analyses for the interim wet storage of

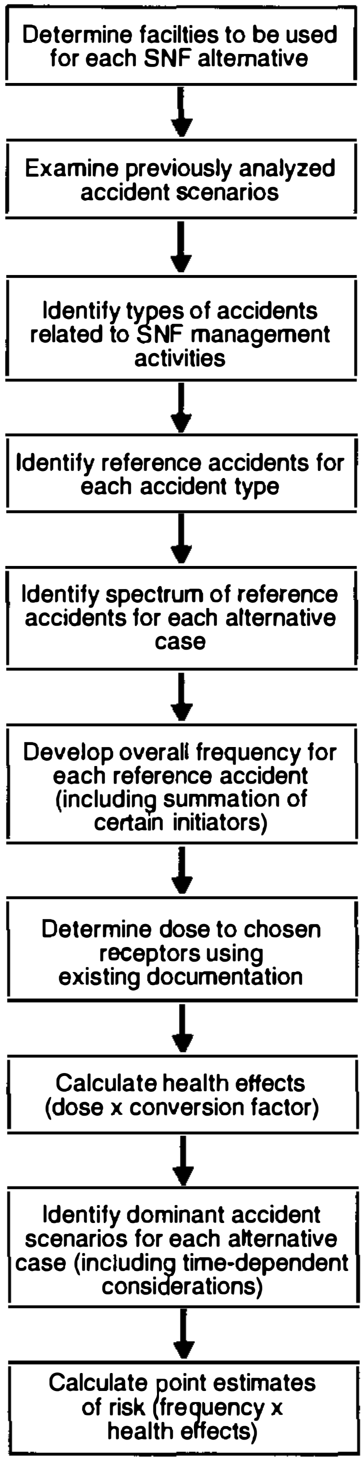


Figure 5-1. Accident analysis process.

spent nuclear fuel at SRS facilities. As indicated in Attachment A, Table A-3, the facilities required under this alternative would consist of existing facilities, including necessary upgrades to support safe interim wet storage. In addition, Attachment A, Table A-4, provides a reference accident spectrum associated with these facilities for this alternative. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this alternative, as well as their estimated frequencies. Table 5-20 lists the accident scenario with the highest point estimates of risk to the general public. Table 5-21 compares the potential radiological accidents and health effects of the interim wet storage (Option 1) of spent nuclear fuel for the No Action alternative.

Table 5-20. Highest point estimates of risk among receptor groups (Option 1).

	Receptor Groups	
	Maximally Exposed	Population to 80 kilometers
Overall Point Estimate of Risk[a]	1.6×10^{-7} (Fuel Assembly Breach)	1.4×10^{-3} (Fuel Assembly Breach)

a. Units of latent fatal cancers per year.

5.15.2.2 Alternative 2 - Decentralization. Accident assessments considered for this alternative include those considered for the No Action alternative for wet storage (Option 2b) plus assessments for the dry storage (Option 2a) of spent nuclear fuel and for the processing of spent fuel (Option 2c). Option 2c (processing) assumes the use of existing facilities to dissolve, separate, and further stabilize spent nuclear fuel. For cases that include some treatment (e.g., canning) of spent nuclear fuel, such treatment is referred to as "stabilization," not processing. The amount of fuel of various types to be considered would include those quantities from the production reactors, existing research fuel, foreign research reactor fuel, and fuel transported for safety or research activities.

5.15.2.2.1 Option 2a - Dry Storage — DOE estimated potential radiological accident impacts that could occur in this case using existing DOE-approved safety analysis reports submitted to DOE by Westinghouse Savannah River Company for vault storage of special nuclear material from existing facilities. DOE has not incorporated the technology to support interim dry storage of spent nuclear fuel at the SRS. To provide a basis for evaluating the potential impacts from this alternative case, this assessment used data from existing safety analyses for special nuclear material storage facilities and extrapolated these data to apply to spent nuclear fuel. DOE also considered radiological accidents associated with wet storage, at least in the near term, because the spent nuclear fuel is currently in wet storage. Similarly, this assessment includes fuel handling accidents throughout the transition phase (i.e., until fuel is in interim dry storage). As indicated in Attachment A, Table A-4,

Table 5-21. Radioactive release accidents and health effects for spent nuclear fuel alternatives.[a,b]

Alternative (by case)	Accident Scenario	Frequency (per year)	Potential Fatal Cancers: Maximally exposed offsite individual[d]	Potential Fatal Cancers: Population to 80 kilometers[d]	Potential Fatal Cancers: Worker[e]	Potential Fatal Cancers: Colocated Worker[e]	Point Estimate of Risk[c]: Maximally exposed offsite individual	Point Estimate of Risk[c]: Population to 80 kilometers[f]	Point Estimate of Risk[c]: Worker	Point Estimate of Risk[c]: Colocated Worker
1. No Action										
Option 1 Wet Storage	A1 Fuel Assembly Breach	$1.6x10^{-1}$	$1.0x10^{-6}$	$8.5x10^{-3}$	(a)	$4.8x10^{-6}$	$1.6x10^{-7}$	$1.4x10^{-3}$	(a)	$7.7x10^{-7}$
	A4 Material Release (Adjacent Facility)	$2.4x10^{-3}$	$3.0x10^{-6}$	$2.5x10^{-2}$	(a)	$2.0x10^{-5}$	$7.2x10^{-9}$	$6.0x10^{-5}$	(a)	$4.8x10^{-8}$
	A5 Criticality in Water	$3.1x10^{-3}$	$1.5x10^{-6}$	$4.4x10^{-3}$	(a)	$5.6x10^{-5}$	$4.7x10^{-9}$	$1.4x10^{-5}$	(a)	$1.7x10^{-7}$
	A7 Spill/Liquid Discharge (external)	$2.0x10^{-4}$	$2.7x10^{-6}$	$9.0x10^{-3}$	(a)	$1.1x10^{-6}$	$5.4x10^{-10}$	$1.8x10^{-6}$	(a)	$2.2x10^{-10}$
	A8 Spill/Liquid Discharge (internal)	$1.1x10^{-1}$	$1.2x10^{-13}$	$1.0x10^{-9}$	(a)	$8.0x10^{-15}$	$1.3x10^{-14}$	$1.1x10^{-10}$	(a)	$8.8x10^{-16}$
2. Decentralization										
Option 2a Dry Storage	A1 Fuel Assembly Breach	$1.6x10^{-1}$	$1.0x10^{-6}$	$8.5x10^{-3}$	(a)	$4.8x10^{-6}$	$1.6x10^{-7}$	$1.4x10^{-3}$	(a)	$7.7x10^{-7}$
	A3 Material Release (Dry Vault)	$1.4x10^{-3}$	$1.1x10^{-9}$	$3.5x10^{-6}$	(a)	(b)	$1.5x10^{-12}$	$4.9x10^{-9}$	(a)	(b)
	A4 Material Release (Adjacent Facility)	$2.4x10^{-3}$	$3.0x10^{-6}$	$2.5x10^{-2}$	(a)	$2.0x10^{-5}$	$7.2x10^{-9}$	$6.0x10^{-5}$	(a)	$4.8x10^{-8}$
	A5 Criticality in Water	$3.1x10^{-3}$	$1.5x10^{-6}$	$4.4x10^{-3}$	(a)	$5.6x10^{-5}$	$4.7x10^{-9}$	$1.4x10^{-5}$	(a)	$1.7x10^{-7}$
	A7 Spill/Liquid Discharge (external)	$2.0x10^{-4}$	$2.7x10^{-6}$	$9.0x10^{-3}$	(a)	$1.1x10^{-6}$	$5.4x10^{-10}$	$1.8x10^{-6}$	(a)	$2.2x10^{-10}$
	A8 Spill/Liquid Discharge (internal)	$1.1x10^{-1}$	$1.2x10^{-13}$	$1.0x10^{-9}$	(a)	$8.0x10^{-15}$	$1.3x10^{-14}$	$1.1x10^{-10}$	(a)	$8.8x10^{-16}$
Option 2b Wet Storage	A1 Fuel Assembly Breach	$1.6x10^{-1}$	$1.0x10^{-6}$	$8.5x10^{-3}$	(a)	$4.8x10^{-6}$	$1.6x10^{-7}$	$1.4x10^{-3}$	(a)	$7.7x10^{-7}$
	A4 Material Release (Adjacent Facility)	$2.4x10^{-3}$	$3.0x10^{-6}$	$2.5x10^{-2}$	(a)	$2.0x10^{-5}$	$7.2x10^{-9}$	$6.0x10^{-5}$	(a)	$4.8x10^{-8}$
	A5 Criticality in Water	$3.1x10^{-3}$	$1.5x10^{-6}$	$4.4x10^{-3}$	(a)	$5.6x10^{-5}$	$4.7x10^{-9}$	$1.4x10^{-5}$	(a)	$1.7x10^{-7}$
	A7 Spill/Liquid Discharge (external)	$2.0x10^{-4}$	$2.7x10^{-6}$	$9.0x10^{-3}$	(a)	$1.1x10^{-6}$	$5.4x10^{-10}$	$1.8x10^{-6}$	(a)	$2.2x10^{-10}$
	A8 Spill/Liquid Discharge (internal)	$1.1x10^{-1}$	$8.2x10^{-13}$	$1.0x10^{-9}$	(a)	$8.0x10^{-15}$	$1.3x10^{-14}$	$1.1x10^{-10}$	(a)	$8.8x10^{-16}$
Option 2c Processing	A1 Fuel Assembly Breach	$1.6x10^{-1}$	$1.0x10^{-6}$	$8.5x10^{-3}$	(a)	$4.8x10^{-6}$	$1.6x10^{-7}$	$1.4x10^{-3}$	(a)	$7.7x10^{-7}$
	A2 Material Release (Processing)	$2.6x10^{-1}$	$3.4x10^{-8}$	$2.6x10^{-4}$	(a)	$3.6x10^{-8}$	$8.9x10^{-9}$	$6.8x10^{-5}$	(a)	$9.4x10^{-9}$

Table 5-21. (continued).

Alternative (by case)	Accident Scenario	Frequency (per year)	Potential Fatal Cancers: Maximally exposed offsite individual[d]	Potential Fatal Cancers: Population to 80 kilometers[d]	Potential Fatal Cancers: Worker[e]	Potential Fatal Cancers: Colocated Worker[e]	Point Estimate of Risk[c]: Maximally exposed offsite individual	Point Estimate of Risk[c]: Population to 80 kilometers[f]	Point Estimate of Risk[c]: Worker	Point Estimate of Risk[c]: Colocated Worker
Option 2c (continued)	A3 Material Release (Dry Vault)	$1.4x10^{-3}$	$1.1x10^{-9}$	$3.5x10^{-6}$	(a)	(b)	$1.5x10^{-12}$	$4.9x10^{-9}$	(a)	(b)
	A4 Material Release (Adjacent Facility)	$2.4x10^{-3}$	$3.0x10^{-6}$	$2.5x10^{-2}$	(a)	$2.0x10^{-5}$	$7.2x10^{-9}$	$6.0x10^{-5}$	(a)	$4.8x10^{-8}$
	A5 Criticality in Water	$3.1x10^{-3}$	$1.5x10^{-6}$	$4.4x10^{-3}$	(a)	$5.6x10^{-5}$	$4.7x10^{-9}$	$1.4x10^{-5}$	(a)	$1.7x10^{-7}$
	A6 Criticality in Processing	$1.4x10^{-4}$	$3.5x10^{-6}$	$4.3x10^{-3}$	(a)	$1.0x10^{-4}$	$4.9x10^{-10}$	$6.0x10^{-7}$	(a)	$1.4x10^{-8}$
	A7 Spill/Liquid Discharge (external)	$2.0x10^{-4}$	$2.7x10^{-6}$	$9.0x10^{-3}$	(a)	$1.1x10^{-6}$	$5.4x10^{-10}$	$1.8x10^{-6}$	(a)	$2.2x10^{-10}$
	A8 Spill/Liquid Discharge (internal)	$1.1x10^{-3}$	$1.2x10^{-13}$	$1.0x10^{-9}$	(a)	$8.0x10^{-15}$	$1.3x10^{-14}$	$1.1x10^{-10}$	(a)	$8.8x10^{-16}$
3. 1992/1993 Planning Basis										
Option 3a Dry Storage	Same as Option 2a for Decentralization									
Option 3b Wet Storage	Same as Option 2b for Decentralization									
Option 3c Processing	Same as Option 2c for Decentralization									
4. Regionalization - A										
Option 4a Dry Storage	Same as Option 2a for Decentralization									
Option 4b Wet Storage	Same as Option 2b for Decentralization									
Option 4c Processing	Same as Option 2c for Decentralization									

Table 5-21. (continued).

Alternative (by case)	Accident Scenario	Frequency (per year)	Potential Fatal Cancers: Maximally exposed offsite individual[d]	Potential Fatal Cancers: Population to 80 kilometers[d]	Potential Fatal Cancers: Worker[e]	Potential Fatal Cancers: Colocated Worker[e]	Point Estimate of Risk[c]: Maximally exposed offsite individual	Point Estimate of Risk[c]: Population to 80 kilometers[f]	Point Estimate of Risk[c]: Worker	Point Estimate of Risk[c]: Colocated Worker
4. Regionalization - B										
Option 4d Dry Storage				Same as Option 2a for Decentralization						
Option 4e Wet Storage				Same as Option 2b for Decentralization						
Option 4f Processing				Same as Option 2c for Decentralization						
Option 4g Shipping Out				Same as Option 1 for No Action						
5. Centralization										
Option 5a Dry Storage				Same as Option 2a for Decentralization						
Option 5b Wet Storage				Same as Option 2b for Decentralization						
Option 5c Processing				Same as Option 2c for Decentralization						
Option 5d Shipping Out				Same as Option 1 No Action						

a. The safety analysis reports from which information was extracted for these accidents were written before the issuance of DOE Order 5480.23; previous Orders did not require the inclusion of workers.

b. The safety analysis reports from which information was extracted for these accidents were written before the issuance of DOE Order 5480.23; previous Orders did not require the inclusion of colocated workers.

c. Units for point estimates of risk are given in potential latent fatal cancers per year.

d. ICRP 60 risk factor for the general public (5.0×10^{-4} fatal cancer per year) was used to determine potential latent fatal cancers.

e. ICRP 60 risk factor for workers (4.0×10^{-4} fatal cancer per year) was used to determine potential latent fatal cancers.

the facilities required under this alternative would consist of existing and new facilities necessary to support the safe handling, stabilization, and dry storage of spent nuclear fuel. In addition, Table A-4 identifies a potential accident spectrum associated with these facilities for this case. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this alternative case, as well as the estimated frequency of occurrence for each accident. Table 5-21 lists the potential radiological accidents and health effects associated with dry storage of spent nuclear fuel for the Decentralization alternative. For the transition period of wet to dry storage, Table 5-22 lists the accident scenario with the highest overall point estimate of risk to the general public. Table 5-22 lists the accident scenario with the highest point estimate of risk (after transition) to the general public when the fuel had been moved from wet storage (after approximately 15 years) and placed in interim dry storage. This indicates a substantial reduction in risk (more than six orders of magnitude) when fuel handling events are no longer potential accident initiators.

Table 5-22. Highest point estimates of risk among receptor groups (Option 2a).

	Receptor Groups	
	Maximally Exposed Offsite Individual	Population to 80 kilometers
Overall Point Estimate of Risk[a]	$1.6x10^{-7}$ (Fuel Assembly Breach)	$1.4x10^{-3}$ (Fuel Assembly Breach)
Transitioned to Dry Storage Point Estimate of Risk[a]	$1.5x10^{-12}$ (Dry Vault Material Release)	$4.9x10^{-9}$ (Dry Vault Material Release)

a. Units of latent fatal cancers per year.

5.15.2.2.2 Option 2b - Wet Storage — DOE estimated potential radiological accident impacts that could occur under this case using existing DOE-approved safety analysis reports and amendments submitted to DOE by Westinghouse Savannah River Company for existing wet storage facilities. As indicated in Attachment A, Table A-4, the facilities (modules as defined in the WSRC 1994b and Figure 3-2) would consist of existing facilities and specific upgrades necessary to support safe interim wet storage. In addition, Table A-4 identifies the reference accident spectrum associated with these facilities for this option. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this alternative option, as well as the estimated frequency of occurrence for each accident. Table 5-21 lists the radiological accidents and consequences of the wet storage (Option 2b) of spent nuclear fuel for the Decentralization alternative. Table 5-23 lists the accident scenario with the highest point estimate of risk to the general public. For wet pool storage options, there are no transition phases.

Table 5-23. Highest point estimates of risk among receptor groups (Option 2b).

	Receptor Groups	
	Maximally Exposed Offsite Individual	Population to 80 kilometers
Overall Point Estimate of Risk[a]	1.6×10^{-7} (Fuel Assembly Breach)	1.4×10^{-3} (Fuel Assembly Breach)

a. Units of latent fatal cancers per year.

5.15.2.2.3 Option 2c - Processing and Storage — Processing for the SRS is defined as the operation of the separations facilities in F- or H-Areas. The H-Area facilities were designed to recover uranium and plutonium from spent production reactor fuel, and the F-Area facilities were designed to recover plutonium.

DOE estimated potential radiological accident impacts that could occur under this option using existing DOE-approved safety analysis reports submitted to DOE by Westinghouse Savannah River Company for processes and for vault storage of special nuclear material from existing facilities. DOE also considered radiological accidents associated with wet storage, because the spent nuclear fuel is currently in wet storage. Similarly, it included fuel handling accidents throughout the processing phase (i.e., until special nuclear material is in interim dry storage). As indicated in Attachment A, Table A-4, the facilities required under this option would consist of existing and new facilities necessary to support safe handling and processing of spent nuclear fuel into special nuclear material for dry storage. In addition, Table A-4 identifies the reference accident spectrum associated with these facilities for this case. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this alternative case, as well as the estimated frequency of occurrence for each accident. Table 5-21 lists the radiological release accidents and health effects for the processing of spent nuclear fuel to special nuclear material for the Decentralization alternative. Table 5-24 lists the accident scenario with the highest overall point estimate of risk to the general public from the transition period of wet spent fuel storage into processing for special nuclear material. When the fuel had been processed from wet storage to special nuclear material and placed in its interim dry storage, Table 5-24 lists the accident scenario with the highest point estimate of risk after transition to the general public. This indicates a substantial reduction in risk (more than six orders of magnitude) when fuel handling events and processing events are no longer potential accident initiators.

Table 5-24. Highest point estimates of risk among receptor groups (Option 2c).

	Receptor Groups	
	Maximally Exposed Offsite Individual	Population to 80 kilometers
Overall Point Estimate of Risk[a]	1.6×10^{-7} (Fuel Assembly Breach)	1.4×10^{-3} (Fuel Assembly Breach)
Transitioned to Dry Storage Point Estimate of Risk[a]	1.5×10^{-12} (Dry Vault Material Release)	4.9×10^{-9} (Dry Vault Material Release)

a. Units of latent fatal cancers per year.

For this option, DOE assumes it could not process some fuel clad in stainless steel or zirconium into special nuclear material and, therefore, would dry-store it as fuel. The technology for dry storage of nonaluminum-clad fuel has been demonstrated and is assumed to pose no greater risk than monitored dry storage of special nuclear material.

5.15.2.3 Alternative 3 - 1992/1993 Planning Basis. Because this alternative would be consistent with the *status quo* at the SRS, existing documents contain sufficient information to examine its accident analysis impacts. The SRS would continue to receive the spent nuclear fuel designated for the Site, and DOE would complete facilities already planned to accommodate the existing inventory and the spent nuclear fuel receipts. This alternative would require the same facilities already used to support the cases discussed in the Section 5.15.2.2. The major difference would be the amount of fuel ultimately stored because this alternative assumes the continued receipt of fuel beyond that shipped to the SRS under the Decentralization alternative.

5.15.2.3.1 Option 3a - Dry Storage — DOE estimated potential radiological accident impacts that could occur under this case using existing DOE-approved safety analysis reports for vault storage from existing facilities and the study discussed for Option 2a. DOE also considered radiological accidents associated with wet storage, at least in the near term, because the spent nuclear fuel is currently in wet storage. Similarly, it included fuel handling accidents throughout the transition phase (i.e., until the fuel is in interim dry storage). As indicated in Attachment A, Table A-4, the facilities required under this option would consist of existing and new facilities necessary to support the safe handling and stabilization of spent nuclear fuel for dry storage. In addition, Table A-4 identifies the reference accident spectrum associated with these facilities for this case. Attachment A, Table A-2, lists the authorization basis references for the source terms considered in analyzing potential accidents under this option, as well as the estimated frequency of occurrence for each accident. Table 5-21 lists the radiological release accidents and health effects for the dry storage of spent nuclear fuel for the 1992/1993 Planning Basis alternative. For the entire period, the accident scenarios with the highest point estimates of risk to the general public would be the same as those for Option 2a, as listed in Table 5-22.

5.15.2.3.2 Option 3b - Wet Storage — DOE estimated potential radiological accident impacts that could occur under this case using existing DOE-approved safety analysis reports and from amendments submitted to DOE by Westinghouse Savannah River Company for wet storage for existing facilities. As indicated in Attachment A, Table A-4, the facilities required under this option would consist of existing facilities and upgrades necessary to support safe interim wet storage. In addition, Table A-4 identifies the reference accident spectrum associated with these facilities for this option. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this option, as well as the estimated frequency of occurrence for each accident. Table 5-21 lists the radiological release accidents and health effects of the wet storage (Option 3b) of spent nuclear fuel for the 1992/1993 Planning Basis alternative. The accident scenario with the highest point estimate of risk to the general public would be the same as that for Option 2b, as listed in Table 5-23.

5.15.2.3.3 Option 3c - Processing and Storage. Table 5-21 lists the radioactive release accidents and health effects for the processing of spent nuclear fuel for this option. After processing is complete, the accident scenario with the highest point estimate of risk would be associated with the storage of special nuclear materials, as discussed for Option 2c and listed in Table 5-24.

5.15.2.4 Alternative 4 - Regionalization. This alternative comprises Regionalization A and Regionalization B subalternatives. Under the Regionalization A subalternative (Options 4a, 4b, and 4c), the SRS would receive all aluminum-clad fuel from the other sites considered in this EIS and would transfer its existing inventory of stainless steel- and Zircaloy-clad fuel to other DOE sites, as appropriate. These proposed activities would reflect current and past activities, so sufficient information and analyses are available to enable the scaling or other extrapolation of radiological accident impacts. The total amount of spent nuclear fuel to be managed under Regionalization A would be slightly less than that for Alternatives 2 and 3; the decisionmaker could use this amount to adjust the estimated point estimate of risk by the use of an appropriate adjustment (scaling) factor, as discussed in Attachment A, Section A.2.9.

Under the Regionalization B subalternative (Options 4d, 4e, 4f, and 4g), the SRS would receive all existing and new spent nuclear fuel east of the Mississippi River. The decisionmaker could use the change in spent nuclear fuel inventories to adjust the estimated point estimate of risk by the use of an appropriate adjustment (scaling) factor, as discussed in Attachment A, Section A.2.9. For the purposes of this evaluation, Option 4g (Section 5.15.2.4.7) assumes that DOE would ship all fuel off the Site to the Oak Ridge Reservation.

5.15.2.4.1 Option 4a - Dry Storage — This case is similar to Option 2a, with the exception of the quantity and type of fuel to be stored. As with Option 2a, this assessment evaluated existing analyses; the point estimates of risk are the same as those for Option 2a.

5.15.2.4.2 Option 4b - Wet Storage — This case is similar to Option 2b, with the exception of a slightly smaller quantity of fuel to be stored. As with Option 2b, this assessment evaluated existing analyses, and the point estimates of risk are the same as those for Option 2b.

5.15.2.4.3 Option 4c - Processing and Storage — For this option, the accident analysis evaluation is similar to Option 2c. DOE assumes that it could process spent nuclear fuel associated with regionalization at SRS with existing facilities, because they are designed to process aluminum-clad fuel. However, the small amount of aluminum-clad fuel received after major processing options are completed would be placed in wet storage.

5.15.2.4.4 Option 4d - Dry Storage — The accident analysis evaluation for this option is similar to that for Option 2a, with the exception of the increased inventories and types of fuel to be stored.

5.15.2.4.5 Option 4e - Wet Storage — The accident analysis evaluation for this option is similar to that for Option 2b, with the exception of the increased inventories and types of fuel to be stored.

5.15.2.4.6 Option 4f - Processing and Storage — For this option, the accident analysis evaluation is similar to Option 2c. DOE assumes that it could process all the current SRS aluminum-clad spent nuclear fuel with existing facilities. However, all receipts of spent nuclear fuel will be placed in dry storage as discussed for Option 4d.

5.15.2.4.7 Option 4g - Shipping Off Site — This option assumes that DOE would characterize the fuel and ship it all off the Site. Thus, the potential radiological accidents considered are the same as those for Alternative 1.

5.15.2.5 Alternative 5 - Centralization. This alternative for the SRS would involve fuel types and new facilities beyond those considered for any other alternative. For instance, under this alternative, the SRS would receive spent nuclear fuel from the U.S. Navy. One of the new facilities that would be necessary to support this type of spent nuclear fuel is the Expended Core Facility (ECF). Volume 1, Appendix D, includes a detailed accident analyses for this proposed facility using SRS-specific parameters.

This alternative would bound the maximum number of spent nuclear fuel-related accident scenarios that DOE could expect at the SRS, due to the number of new facilities at the Site that would have to accommodate the diversity and the increased amount of the fuel to be managed. The decisionmaker could use this maximum amount of spent nuclear fuel to adjust the estimated risk by the use of an appropriate scaling factor, as discussed in Attachment A, Section A.2.9. For the purposes of this evaluation, Option 5d (Section 5.15.2.5.4) assumes that DOE would ship all fuel off the Site to another DOE facility.

5.15.2.5.1 Option 5a - Dry Storage — The major difference in dry storage facilities between this alternative and the others would be the addition of a facility for Naval spent nuclear fuels and the large quantity of spent fuel shipped to the SRS from the Hanford Site. DOE estimated potential radiological accident impacts that could occur under this option using DOE-approved safety analysis reports submitted to DOE by Westinghouse Savannah River Company for vault storage in existing facilities at the SRS and the study discussed for Option 2a. In addition, DOE considered radiological accidents associated with wet storage, at least in the near term, because the SRS spent nuclear fuel is currently in wet storage. Similarly, it included fuel handling accidents throughout the transition phase (i.e., until fuel is in interim dry storage). As indicated in Attachment A, Table A-4, the facilities required under this option would consist of existing and new facilities necessary to support the safe handling and stabilization of spent nuclear fuel for dry storage. In addition, Table A-4 identifies the reference accident spectrum associated with these facilities for this case. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this option, as well as the estimated frequency of occurrence for each accident. Table 5-21 compares the radiological release accidents and health effects for the dry storage of spent nuclear fuel for the Centralization alternative. From the transition period of wet to dry storage, the accident scenario with the highest point estimate of risk to the general public would be the same as that for Option 2a, as listed in Table 5-22. When the fuel had been moved from wet storage (after approximately 25 years) and placed in interim dry storage, the accident scenario with the highest point estimate of risk to the population would be the same as the Option 2a dry storage phase.

5.15.2.5.2 Option 5b - Wet Storage — The accident analysis evaluation for this option is similar to that for Option 2b, with the exception of the amount and type of fuel to be stored.

5.15.2.5.3 Option 5c - Processing and Storage — For this option, the accident analysis evaluation is similar to Option 2c. DOE assumes that it could process the current SRS aluminum-clad spent nuclear fuel with existing facilities. However, the SRS would place all receipts of fuel in dry storage, as discussed for Option 5a.

5.15.2.5.4 Option 5d - Shipping Off Site — This option assumes that DOE would perform the characterization of the fuel at the SRS, and then would ship all fuel off the Site. Thus, the potential radiological accidents considered are the same as those for the No Action alternative.

5.15.3 Chemical Hazard Evaluation

For toxic chemicals, several government agencies recommend the quantification of health effects as threshold values of concentrations in air or water that cause short-term effects. The long-term health consequences of human exposure to toxic chemicals are not as well understood as those for radiation. Thus, the potential health effects from toxic chemicals are more subjective than those from radioactive materials.

This section provides a quantitative discussion for an analyzed chemical accident at the Receiving Basin for Offsite Fuel facility and qualitative discussions addressing chemical hazards for each of the other existing SRS facilities involved in the receipt, processing, transport, or storage of spent nuclear fuel.

5.15.3.1 Receiving Basin for Offsite Fuel. The maximum reasonably foreseeable chemical hazard accident for the Receiving Basin for Offsite Fuel would involve the release of nitrogen dioxide vapor following the complete reaction of a drum of target cleaning solution (13.4 percent nitric acid) with sodium nitrite (WSRC 1993b). The initiator for this accident is a leak from a storage tank into the target cleaning solution and involves multiple failures or maloperations with an accident probability comparable to that of a natural phenomena accident. Table 5-25 shows the concentration of nitrogen dioxide vapor that an individual at the SRS boundary and a maximally exposed colocated worker could receive.

Table 5-25. Results of analyzed chemical accident.

Receptor Group	Frequency (per year)	NO_2 Concentration (mg/m^3)
Maximally Exposed Offsite Individual	1.0×10^{-3}	0.083
Colocated Worker	1.0×10^{-3}	0.64

To determine the potential health effects from this bounding chemical accident scenario, this assessment was to compare the resulting airborne concentrations of nitrogen dioxide at various receptor distances against Emergency Response Planning Guideline (ERPG) values, where available. Because there were no ERPG values available for nitrogen dioxide, the assessment substituted other chemical toxicity values as follows:

- For Emergency Response Planning Guideline 1, the assessment substituted threshold limit values/time-weighted average (TLV/TWA) values (ACGIH 1987). The time-weighted average is the average concentration for a normal 8-hour workday and a 40-hour workweek from which nearly all workers could receive repeated exposure, day-after-day, without adverse effect.

- For Emergency Response Planning Guideline 2, the assessment substituted level of concern (LOC) values [equal to 0.1 of the immediately dangerous to life or health (IDLH) value; - see below]. The level of concern value is the concentration of a hazardous substance in the air above which there could be serious irreversible health effects or death as a result of a single exposure for a relatively short period of time (EPA 1987).

- For Emergency Response Planning Guideline 3, the assessment substituted immediately dangerous to life or health values. This value is the maximum concentration from which a person could escape within 30 minutes without a respirator and without experiencing any impairment of escape or irreversible side effects (NIOSH 1990).

These values as they apply to nitrogen dioxide are as follows:

- Time-weighted average value = 5.6 milligrams per cubic meter
- Level of concern value = 9.4 milligrams per cubic meter
- Immediately dangerous to life or health value = 94.0 milligrams per cubic meter

5.15.3.2 Reactor Basins. There are no postulated chemical accidents for the reactor basins that would cause an impact to an individual at the SRS boundary or a colocated worker.

5.15.3.3 H-Area. There are no postulated chemical accidents for the H-Area Canyon that would cause an impact to an individual at the SRS boundary or a colocated worker. DOE has performed an accident analysis for the H-Area Canyon facility workers that indicates the existence of potential injuries due to chemical contamination or exposure to hazardous vapors at or above the level of concern exposure limit (Du Pont 1983a). The analysis does not project exposure to hazardous vapors at or above the immediate danger to life and health level to occur.

The probability that a worker could be accidentally exposed to any of the hazardous liquids identified in Attachment A, Table A-14, is bounded by a frequency of 2.8×10^{0} per year (Du Pont 1983a). The most likely injury is an acid burn to the skin.

The probability for exposure to hazardous vapors at or above the level of concern exposure limit is 8.5×10^{-1} per year (Du Pont 1983a). The potential for chemical uptakes and for illness would depend on the safety measures taken before the exposure, the duration of the exposure, and the mitigating actions taken after the exposure.

5.15.3.4 F-Area. There are no postulated chemical accidents for the F-Area Canyon that would cause an impact to an individual at the SRS boundary or a colocated worker. DOE has performed an accident analysis for the F-Area Canyon facility workers that indicates the existence of potential injuries due to chemical contamination or exposure to hazardous vapors at or above the level of concern exposure limit (Du Pont 1983b). The analysis does not project exposure to hazardous vapors at or above the immediate danger to life and health level to occur.

The probability that a worker could be accidentally exposed to any one of the hazardous liquids identified in Attachment A, Table A-15, is bounded by a frequency of 1.2×10^{0} per year (Du Pont 1983b). The most likely injury is an acid burn to the skin.

The probability for exposure to hazardous vapors at or above the level of concern exposure limit is 3.2×10^{-1} per year (Du Pont 1983b). The potential for chemical uptakes and for illness would depend on the safety measures taken before the exposure, the duration of the exposure, and the mitigating actions taken after the exposure.

5.15.4 Secondary Impacts

The primary focus of the accident analysis is to determine the magnitude of the consequences of postulated accident scenarios on public and worker health and safety. However, DOE recognizes that chemical and radiological accidents can also adversely affect the surrounding environment (i.e., secondary impacts). Accordingly, DOE has qualitatively evaluated each of the eight radiological accident scenarios considered in this analysis for potential secondary impacts. The following paragraphs discuss the results of the evaluation, and Table 5-26 summarizes expected secondary impacts for each accident scenario.

5.15.4.1 Biotic Resources. With the exception of a direct discharge of disassembly basin water to an onsite stream, DOE does not expect radiological contamination resulting from any of the analyzed accidents to reach any onsite or offsite surface water. DOE previously evaluated the case of a direct discharge of disassembly basin water (DOE 1990) and believes that impacts on biotic resources would be minor. Therefore, the impacts on aquatic biota from any of the accident scenarios would be minor. Small areas of minor surface contamination likely would be outside the industrialized area of a postulated accident. Terrestrial biota in or near the contaminated area would be exposed to small quantities of radioactive materials and ionizing radiation until the affected area could be decontaminated. DOE believes that the impacts on biotic resources from this exposure would be minor.

5.15.4.2 Water Resources. DOE expects no adverse impacts on water quality from any of the postulated accident scenarios. Accident A7 (External Spill/Liquid Discharge) would be expected to have the most significant impact. With the exception of the reactor disassembly basins, the location and configuration of existing or potential facilities would prevent a direct release of radionuclide-contaminated water to surface water. However, contamination of the surface aquifer in the area of the release would be likely. The processes governing the slow plume movement and attenuation of contaminants described in Section 5.8 would prevent the contamination from reaching surface- or groundwater resources. Similarly, radionuclide contamination of onsite or offsite drinking

Table 5-26. Qualitative summary of expected secondary impacts.

		Environmental or social factor							
Accident Scenario	Accident Description	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
A1	Fuel assembly breach	No adverse effects on biota expected.	No adverse effects expected to surface or groundwater resources.	Limited economic impacts are expected. Any required cleanup could be handled with existing workforce.	No effect.	Local contamination expected around site of the accident. Minor contamination outside the immediate facility area unlikely to require cleanup of more than 10 acres.	No impacts expected.	No change expected. No irreversible impacts.	No impact to Native American or public lands expected.
A2	Material release (processing)	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A3	Material release (dry vault)	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A4	Material release (adjacent facility)	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A5	Criticality in water	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A6	Criticality during processing	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A7	External spill/liquid discharge	Same as A1.	Surface-water table contamination expected in area of the release. No adverse effects expected to surface-water or drinking water aquifers.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A8	Internal spill/liquid discharge	Same as A1.	No adverse impact to water resources. The spill is expected to be contained entirely within the building structure.	Same as A1.	Same as A1.	Limited contamination is expected outside the effected building.	Same as A1.	Same as A1.	Same as A1.

water sources would be unlikely. DOE evaluated the effects of a direct discharge of disassembly basin water on water resources (DOE 1990) and believes that impacts on water resources would be minimal.

5.15.4.3 Economic Impacts. DOE expects limited economic impacts as a result of any of the postulated accidents. Any cleanup required would be localized, and the existing workforce and equipment could perform it. Contamination should be contained within a small area inside the SRS boundaries for all eight postulated accident scenarios. The existing workforce could accomplish any required cleanup.

5.15.4.4 National Defense. None of the postulated accidents would affect the DOE national defense mission. Spent nuclear fuel management activities do not involve the production of materials needed for national defense.

5.15.4.5 Environmental Contamination. DOE expects that none of the postulated accident scenarios would result in large areas of contamination. Local contamination is likely around the site of an accident, but in all scenarios should be contained within the SRS boundaries. Minor contamination outside the immediate area of the accident is unlikely to require cleanup of more than a small area inside the Site boundary. Impacts in all cases should be minimal.

5.15.4.6 Endangered Species. There are no Federally listed threatened or endangered species habitats in the immediate vicinity of existing or potential spent nuclear fuel storage or processing facilities (see Section 4.9.4). None of the postulated accident scenarios would likely result in large areas of surface contamination outside the immediate facilities, and DOE does not expect adverse impacts to surface water. Therefore, none of the postulated accident scenarios is likely to impact threatened or endangered species.

5.15.4.7 Land Use. No accident scenario should result in large areas of contamination, nor would the impacts be irreversible. DOE expects no change in land use.

5.15.4.8 Treaty Rights. The environmental impacts of each of the accident scenarios should be contained within the SRS boundaries. Because there are no Native American or public lands within the site boundaries, treaty rights would not be affected.

5.15.5 Adjusted Point Estimate of Risk Summary

The accident scenarios described in Section 5.15.2 differ only slightly between the various alternatives. These scenarios did not account for variations in spent nuclear fuel shipments (including onsite operational transfers) and spent fuel storage inventories across the alternatives. To provide a realistic comparison across alternatives, DOE developed adjustment factors to adjust frequencies or consequences, depending on the specific circumstance of each alternative. Attachment A, Section A.2.9, provides the methodology and justifications used to develop appropriate adjustment factors. This section provides the adjusted point estimates of risk for each accident scenario by receptor group to demonstrate a relative comparison of each alternative on a case-by-case basis. Tables 5-27, 5-28, and 5-29 summarize the adjusted point estimates of risk for each alternative for the maximally exposed individual, the general population to 80 kilometers, and the colocated worker.

5.16 Cumulative Impacts

The Savannah River Site (SRS) contains major U.S. Department of Energy (DOE) and non-DOE facilities, unrelated to spent nuclear fuel, that would continue to operate throughout the life of the spent nuclear fuel management program. The activities associated with these existing facilities produce environmental consequences that this document has included in the baseline environmental conditions (Chapter 4) against which it assesses the consequences of the spent nuclear fuel alternatives. Impacts of both the construction and operation of SRS spent nuclear fuel facilities would be cumulative with the impacts of existing and planned facilities unrelated to spent nuclear fuel.

This cumulative impact assessment considered the incremental and synergistic effects of the operation of the Defense Waste Processing Facility, which is nearing completion, and the Consolidated Incineration Facility, which is under construction, when appropriate and when data existed. For example, the Air Quality analysis factored in emissions from these two facilities when considering potential impacts of operations of spent nuclear fuel facilities. The small volumes of liquid effluent (treated sanitary wastes) currently entering the environment from the Defense Waste Processing Facility, on the other hand, were considered part of the Water Quality baseline. The only major stand alone facilities scheduled to be built in the near future on the SRS are the Savannah River Ecology Laboratory Conference Center and the new Centralized Sanitary Wastewater Treatment Facility. A number of other planned facilities have not been factored into the cumulative impacts analysis because final funding approval has not been received or because decisions on these facilities involve major

Table 5-27. Adjusted point estimates of risk for the maximally exposed offsite individual (radiological accidents).

		No Action	Decentralization			92/93 Planning Basis			Regionalization - A			Centralization			
Accident Description	Attribute[d]	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c	Option 5a	Option 5b	Option 5c	Option 5d
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$	$1.0x10^{-6}$
	Adjusted Annual Frequency	$1.6x10^{-1}$	$3.3x10^{-1}$	$3.5x10^{-1}$	$1.6x10^{-1}$	$4.0x10^{-1}$	$4.0x10^{-1}$	$2.3x10^{-1}$	$4.4x10^{-1}$	$4.4x10^{-1}$	$2.8x10^{-1}$	$8.4x10^{-1}$	$8.4x10^{-1}$	$6.8x10^{-1}$	$1.7x10^{-1}$
	Adjusted Point Estimate of Risk[b]	$1.6x10^{-7}$	$3.3x10^{-7}$	$3.5x10^{-7}$	$1.6x10^{-7}$	$4.0x10^{-7}$	$4.0x10^{-7}$	$2.3x10^{-7}$	$4.4x10^{-7}$	$4.4x10^{-7}$	$2.8x10^{-7}$	$8.4x10^{-7}$	$8.4x10^{-7}$	$6.8x10^{-7}$	$1.7x10^{-7}$
A2 - Processing release	Adjusted Health Effects[a]	(c)	(c)	(c)	$3.4x10^{-8}$	(c)	(c)	$3.4x10^{-8}$	(c)	(c)	$3.4x10^{-8}$	(c)	(c)	$3.4x10^{-6}$	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	$2.7x10^{-1}$	(c)	(c)	$2.7x10^{-1}$	(c)	(c)	$2.7x10^{-1}$	(c)	(c)	$3.5x10^{0}$	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	$9.2x10^{-9}$	(c)	(c)	$9.2x10^{-9}$	(c)	(c)	$9.2x10^{-9}$	(c)	(c)	$1.2x10^{-7}$	(c)
A3 - Dry vault release	Adjusted Health Effects[a]	(c)	$1.1x10^{-9}$	(c)	$1.1x10^{-9}$	$1.2x10^{-9}$	(c)	$1.2x10^{-9}$	$1.1x10^{-9}$	(c)	$1.1x10^{-9}$	$1.5x10^{-8}$	(c)	$1.5x10^{-8}$	(c)
	Adjusted Annual Frequency	(c)	$1.4x10^{-3}$	(c)	$1.4x10^{-3}$	$1.4x10^{-3}$	(c)	$1.4x10^{-3}$	$1.4x10^{-3}$	(c)	$1.4x10^{-3}$	$1.4x10^{-3}$	(c)	$1.4x10^{-3}$	(c)
	Adjusted Point Estimate of Risk[b]	(c)	$1.6x10^{-12}$	(c)	$1.6x10^{-12}$	$1.6x10^{-12}$	(c)	$1.6x10^{-12}$	$1.5x10^{-12}$	(c)	$1.5x10^{-12}$	$2.1x10^{-11}$	(c)	$2.1x10^{-11}$	(c)
A4 - Adjacent facility release	Adjusted Health Effects[a]	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$	$3.0x10^{-6}$
	Adjusted Annual Frequency	$2.4x10^{-3}$	$5.0x10^{-3}$	$5.3x10^{-3}$	$2.5x10^{-3}$	$5.9x10^{-3}$	$5.9x10^{-3}$	$3.4x10^{-3}$	$6.6x10^{-3}$	$6.6x10^{-3}$	$4.2x10^{-3}$	$1.3x10^{-2}$	$1.3x10^{-2}$	$1.0x10^{-2}$	$2.5x10^{-3}$
	Adjusted Point Estimate of Risk[b]	$7.2x10^{-9}$	$1.5x10^{-8}$	$1.6x10^{-8}$	$7.4x10^{-8}$	$1.8x10^{-8}$	$1.8x10^{-8}$	$1.0x10^{-8}$	$2.0x10^{-8}$	$2.0x10^{-8}$	$1.3x10^{-8}$	$3.8x10^{-8}$	$3.8x10^{-8}$	$3.0x10^{-8}$	$7.4x10^{-9}$

Table 5-27. (continued).

		No Action	Decentralization			92/93 Planning Basis			Regionalization - A			Centralization			
Accident Description	Attribute[a]	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c	Option 5a	Option 5b	Option 5c	Option 5d
A5 - Criticality in water	Adjusted Health Effect[e]	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$
	Adjusted Annual Frequency	$3.1x10^{-3}$	$6.4x10^{-3}$	$6.8x10^{-3}$	$3.2x10^{-3}$	$7.7x10^{-3}$	$7.7x10^{-3}$	$4.4x10^{-3}$	$8.6x10^{-3}$	$8.6x10^{-3}$	$5.5x10^{-3}$	$1.6x10^{-2}$	$1.6x10^{-2}$	$1.3x10^{-2}$	$3.3x10^{-3}$
	Adjusted Point Estimate of Risk[b]	$4.7x10^{-9}$	$9.7x10^{-9}$	$1.0x10^{-8}$	$4.8x10^{-9}$	$1.2x10^{-8}$	$1.2x10^{-8}$	$6.7x10^{-9}$	$1.3x10^{-8}$	$1.3x10^{-8}$	$8.3x10^{-9}$	$2.5x10^{-8}$	$2.5x10^{-8}$	$2.0x10^{-8}$	$5.0x10^{-9}$
A6 - Criticality during processing	Adjusted Health Effects[e]	(c)	(c)	(c)	$3.5x10^{-6}$	(c)	(c)	$3.5x10^{-6}$	(c)	(c)	$3.5x10^{-6}$	(c)	(c)	$3.5x10^{-6}$	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	$1.5x10^{-4}$	(c)	(c)	$1.5x10^{-4}$	(c)	(c)	$1.4x10^{-4}$	(c)	(c)	$1.9x10^{-3}$	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	$5.3x10^{-10}$	(c)	(c)	$5.3x10^{-10}$	(c)	(c)	$4.9x10^{-10}$	(c)	(c)	$6.6x10^{-9}$	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[e]	$2.7x10^{-6}$	$2.8x10^{-6}$	$2.8x10^{-6}$	$2.8x10^{-6}$	$2.8x10^{-6}$	$2.8x10^{-6}$	$2.8x10^{-6}$	$2.8x10^{-6}$	$2.8x10^{-6}$	$2.8x10^{-6}$	$3.8x10^{-5}$	$3.8x10^{-5}$	$3.8x10^{-5}$	$3.8x10^{-5}$
	Adjusted Annual Frequency	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$
	Adjusted Point Estimate of Risk[b]	$5.4x10^{-10}$	$5.4x10^{-10}$	$5.4x10^{-10}$	$5.4x10^{-10}$	$5.4x10^{-10}$	$5.4x10^{-10}$	$5.4x10^{-10}$	$5.4x10^{-10}$	$5.4x10^{-10}$	$5.4x10^{-10}$	$7.6x10^{-9}$	$7.6x10^{-9}$	$7.6x10^{-9}$	$7.6x10^{-9}$
A8 - Internal spill/liquid discharge	Adjusted Health Effects[e]	$1.2x10^{-13}$	$1.2x10^{-13}$	$1.2x10^{-13}$	$1.2x10^{-13}$	$1.3x10^{-13}$	$1.3x10^{-13}$	$1.3x10^{-13}$	$1.2x10^{-13}$	$1.2x10^{-13}$	$1.2x10^{-13}$	$1.6x10^{-12}$	$1.6x10^{-12}$	$1.6x10^{-12}$	$1.6x10^{-12}$
	Adjusted Annual Frequency	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$
	Adjusted Point Estimate of Risk[b]	$1.3x10^{-14}$	$1.3x10^{-14}$	$1.3x10^{-14}$	$1.3x10^{-14}$	$1.4x10^{-14}$	$1.4x10^{-14}$	$1.4x10^{-14}$	$1.3x10^{-14}$	$1.3x10^{-14}$	$1.3x10^{-14}$	$1.8x10^{-13}$	$1.8x10^{-13}$	$1.8x10^{-13}$	$1.3x10^{-14}$

Table 5-27. (continued).

		Regionalization - B			
Accident Description	Attribute[a]	Option 4d	Option 4e	Option 4f	Option 4g
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}
	Adjusted Annual Frequency	4.1×10^{-1}	4.1×10^{-1}	2.5×10^{-1}	1.7×10^{-1}
	Adjusted Point Estimate of Risk[b]	4.1×10^{-7}	4.1×10^{-7}	2.5×10^{-7}	1.7×10^{-7}
A2 - Processing release	Adjusted Health Effects[a]	(c)	(c)	3.4×10^{-8}	(c)
	Adjusted Annual Frequency	(c)	(c)	3.4×10^{-1}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	1.2×10^{-8}	(c)
A3 - Dry vault release	Adjusted Health Effects[a]	1.4×10^{-9}	(c)	1.4×10^{-9}	(c)
	Adjusted Annual Frequency	1.4×10^{-3}	(c)	1.4×10^{-3}	(c)
	Adjusted Point Estimate of Risk[b]	2.0×10^{-12}	(c)	2.0×10^{-12}	(c)
A4 - Adjacent facility release	Adjusted Health Effects[a]	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}
	Adjusted Annual Frequency	6.2×10^{-3}	6.2×10^{-3}	3.7×10^{-3}	2.5×10^{-3}
	Adjusted Point Estimate of Risk[b]	1.9×10^{-8}	1.9×10^{-8}	1.1×10^{-8}	7.5×10^{-9}

Table 5-27. (continued).

Accident Description	Attribute[d]	Regionalization - B Option 4d	Option 4e	Option 4f	Option 4g
A5 - Criticality in water	Adjusted Health Effect[a]	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}
	Adjusted Annual Frequency	8.0×10^{-3}	8.0×10^{-3}	4.8×10^{-3}	3.3×10^{-3}
	Adjusted Point Estimate of Risk[b]	1.2×10^{-8}	1.2×10^{-8}	7.2×10^{-9}	4.9×10^{-9}
A6 - Criticality during processing	Adjusted Health Effects[a]	(c)	(c)	3.5×10^{-6}	(c)
	Adjusted Annual Frequency	(c)	(c)	1.8×10^{-4}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	6.3×10^{-10}	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[a]	3.5×10^{-6}	3.5×10^{-6}	3.5×10^{-6}	3.5×10^{-6}
	Adjusted Annual Frequency	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}
	Adjusted Point Estimate of Risk[b]	7.0×10^{-10}	7.0×10^{-10}	7.0×10^{-10}	7.0×10^{-10}
A8 - Internal spill/liquid discharge	Adjusted Health Effects[a]	1.6×10^{-13}	1.6×10^{-13}	1.6×10^{-13}	1.6×10^{-13}
	Adjusted Annual Frequency	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}
	Adjusted Point Estimate of Risk[b]	1.7×10^{-14}	1.7×10^{-14}	1.7×10^{-14}	1.7×10^{-14}

a. Units for adjusted health effects are given in terms of potential fatal cancers.
b. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
c. The accident scenario is not included in the spectrum of potential accidents for this case.
d. Adjustment factors were calculated using March 1994 data and information. In-process revisions to these data and information should not result in changes to these factors by more than 10 percent.

Table 5-28. Adjusted point estimates of risk for the colocated worker (radiological accidents).

		No Action	Decentralization			92/93 Planning Basis			Regionalization - A			Centralization			
Accident Description	Attribute	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c	Option 5a	Option 5b	Option 5c	Option 5d
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}
	Adjusted Annual Frequency	1.6×10^{-1}	3.3×10^{-1}	3.5×10^{-1}	1.6×10^{-1}	4.0×10^{-1}	4.0×10^{-1}	2.3×10^{-1}	4.4×10^{-1}	4.4×10^{-1}	2.8×10^{-1}	8.4×10^{-1}	8.4×10^{-1}	6.8×10^{-1}	1.7×10^{-1}
	Adjusted Point Estimate of Risk[b]	7.7×10^{-7}	1.6×10^{-6}	1.7×10^{-6}	7.7×10^{-7}	1.9×10^{-6}	1.9×10^{-6}	1.1×10^{-6}	2.1×10^{-6}	2.1×10^{-6}	1.3×10^{-6}	4.0×10^{-6}	4.0×10^{-6}	3.3×10^{-6}	8.2×10^{-7}
A2 - Processing release	Adjusted Health Effects[a]	(c)	(c)	(c)	3.6×10^{-8}	(c)	(c)	3.6×10^{-8}	(c)	(c)	3.6×10^{-8}	(c)	(c)	3.6×10^{-8}	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	2.7×10^{-1}	(c)	(c)	2.7×10^{-1}	(c)	(c)	2.7×10^{-1}	(c)	(c)	3.5×10^{0}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	9.7×10^{-9}	(c)	(c)	9.7×10^{-9}	(c)	(c)	9.7×10^{-9}	(c)	(c)	1.3×10^{-7}	(c)
A3 - Dry vault release	Adjusted Health Effects[a]	(c)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(c)
	Adjusted Annual Frequency	(c)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(c)
A4 - Adjacent facility release	Adjusted Health Effects[a]	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}
	Adjusted Annual Frequency	2.4×10^{-3}	5.0×10^{-3}	5.3×10^{-3}	2.5×10^{-3}	5.9×10^{-3}	5.9×10^{-3}	3.4×10^{-3}	6.6×10^{-3}	6.6×10^{-3}	4.2×10^{-3}	1.3×10^{-2}	1.3×10^{-2}	1.0×10^{-2}	2.5×10^{-3}
	Adjusted Point Estimate of Risk[b]	4.8×10^{-8}	1.0×10^{-7}	1.1×10^{-7}	4.9×10^{-8}	1.2×10^{-7}	1.2×10^{-7}	6.8×10^{-8}	1.3×10^{-7}	1.3×10^{-7}	8.5×10^{-8}	2.5×10^{-7}	2.5×10^{-7}	2.0×10^{-7}	5.0×10^{-8}
A5 - Criticality in water	Adjusted Health Effects[a]	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}
	Adjusted Annual Frequency	3.1×10^{-3}	6.4×10^{-3}	6.8×10^{-3}	3.2×10^{-3}	7.7×10^{-3}	7.7×10^{-3}	4.4×10^{-3}	8.6×10^{-3}	8.6×10^{-3}	5.5×10^{-3}	1.6×10^{-2}	1.6×10^{-2}	1.3×10^{-2}	3.3×10^{-3}
	Adjusted Point Estimate of Risk[b]	1.7×10^{-7}	3.6×10^{-7}	3.8×10^{-7}	1.8×10^{-7}	4.3×10^{-7}	4.3×10^{-7}	2.5×10^{-7}	4.8×10^{-7}	4.8×10^{-7}	3.1×10^{-7}	9.0×10^{-7}	9.0×10^{-7}	7.3×10^{-7}	1.8×10^{-7}

Table 5-28. (continued).

		No Action	Decentralization			92/93 Planning Basis			Regionalization - A			Centralization			
Accident Description	Attribute	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c	Option 5a	Option 5b	Option 5c	Option 5d
A6 - Criticality during processing	Adjusted Health Effects[a]	(c)	(c)	(c)	1.0×10^{-4}	(c)	(c)	1.0×10^{-4}	(c)	(c)	1.0×10^{-4}	(c)	(c)	1.0×10^{-4}	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	1.5×10^{-4}	(c)	(c)	1.5×10^{-4}	(c)	(c)	1.4×10^{-4}	(c)	(c)	1.9×10^{-3}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	1.5×10^{-8}	(c)	(c)	1.5×10^{-8}	(c)	(c)	1.4×10^{-8}	(c)	(c)	1.9×10^{-7}	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[a]	3.0×10^{-5}	3.1×10^{-5}	3.1×10^{-5}	3.1×10^{-5}	3.2×10^{-5}	3.2×10^{-5}	3.2×10^{-5}	3.1×10^{-5}	3.1×10^{-5}	3.1×10^{-5}	4.1×10^{-4}	4.1×10^{-4}	4.1×10^{-4}	4.1×10^{-4}
	Adjusted Annual Frequency	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}
	Adjusted Point Estimate of Risk[b]	6.0×10^{-9}	6.2×10^{-9}	6.2×10^{-9}	6.2×10^{-9}	6.4×10^{-9}	6.4×10^{-9}	6.4×10^{-9}	6.2×10^{-9}	6.2×10^{-9}	6.2×10^{-9}	8.2×10^{-8}	8.2×10^{-8}	8.2×10^{-8}	8.2×10^{-8}
A8 - Internal spill/liquid discharge	Adjusted Health Effects[a]	8.0×10^{-15}	8.3×10^{-15}	8.3×10^{-15}	8.3×10^{-15}	8.4×10^{-15}	8.4×10^{-15}	8.4×10^{-15}	8.2×10^{-15}	8.2×10^{-15}	8.2×10^{-15}	1.1×10^{-13}	1.1×10^{-13}	1.1×10^{-13}	1.1×10^{-13}
	Adjusted Annual Frequency	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}
	Adjusted Point Estimate of Risk[b]	8.8×10^{-16}	9.2×10^{-16}	9.2×10^{-16}	9.2×10^{-16}	9.2×10^{-16}	9.2×10^{-16}	9.2×10^{-16}	9.1×10^{-16}	9.1×10^{-16}	9.1×10^{-16}	1.2×10^{-14}	1.2×10^{-14}	1.2×10^{-14}	1.2×10^{-14}

Table 5-28. (continued).

		Regionalization - B			
Accident Description	Attribute	Option 4d	Option 4e	Option 4f	Option 4g
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	$4.8x10^{-6}$	$4.8x10^{-6}$	$4.8x10^{-6}$	$4.8x10^{-6}$
	Adjusted Annual Frequency	$4.1x10^{-1}$	$4.1x10^{-1}$	$2.5x10^{-1}$	$1.7x10^{-1}$
	Adjusted Point Estimate of Risk[b]	$2.0x10^{-6}$	$2.0x10^{-6}$	$1.2x10^{-6}$	$8.1x10^{-7}$
A2 - Processing release	Adjusted Health Effects[a]	(c)	(c)	$3.6x10^{-8}$	(c)
	Adjusted Annual Frequency	(c)	(c)	$3.4x10^{-1}$	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	$1.2x10^{-8}$	(c)
A3 - Dry vault release	Adjusted Health Effects[a]	(c)	(c)	(d)	(c)
	Adjusted Annual Frequency	(c)	(c)	(d)	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(d)	(c)
A4 - Adjacent facility release	Adjusted Health Effects[a]	$2.0x10^{-5}$	$2.0x10^{-5}$	$2.0x10^{-5}$	$2.0x10^{-5}$
	Adjusted Annual Frequency	$6.2x10^{-3}$	$6.2x10^{-3}$	$3.7x10^{-3}$	$2.5x10^{-3}$
	Adjusted Point Estimate of Risk[b]	$1.2x10^{-7}$	$1.2x10^{-7}$	$7.4x10^{-7}$	$5.0x10^{-8}$

Table 5-28. (continued).

Accident Description	Attribute	Regionalization - B Option 4d	Option 4e	Option 4f	Option 4g
A5 - Criticality in water	Adjusted Health Effects[a]	$5.6x10^{-5}$	$5.6x10^{-5}$	$5.6x10^{-5}$	$5.6x10^{-5}$
	Adjusted Annual Frequency	$8.0x10^{-3}$	$8.0x10^{-3}$	$4.8x10^{-3}$	$3.3x10^{-3}$
	Adjusted Point Estimate of Risk[b]	$4.5x10^{-7}$	$4.5x10^{-7}$	$2.7x10^{-7}$	$1.8x10^{-7}$
A6 - Criticality during processing	Adjusted Health Effects[a]	(c)	(c)	$1.0x10^{-4}$	(c)
	Adjusted Annual Frequency	(c)	(c)	$1.8x10^{-4}$	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	$1.8x10^{-8}$	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[a]	$3.9x10^{-3}$	$3.9x10^{-3}$	$3.9x10^{-3}$	$3.9x10^{-3}$
	Adjusted Annual Frequency	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$
	Adjusted Point Estimate of Risk[b]	$7.8x10^{-7}$	$7.8x10^{-7}$	$7.8x10^{-7}$	$7.8x10^{-7}$
A8 - Internal spill/liquid discharge	Adjusted Health Effects[a]	$1.0x10^{-14}$	$1.0x10^{-14}$	$1.0x10^{-14}$	$1.0x10^{-14}$
	Adjusted Annual Frequency	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$
	Adjusted Point Estimate of Risk[b]	$1.2x10^{-15}$	$1.2x10^{-15}$	$1.2x10^{-15}$	$1.2x10^{-15}$

a. Units for adjusted health effects are given in terms of potential fatal cancers.
b. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
c. The accident scenario is not included in the spectrum of potential accidents for this case.
d. The safety analyses from which information was extracted for these accidents were written before issuance of DOE Order 5480.23; previous Orders did not require the inclusion of colocated workers.

Table 5-29. Adjusted point estimates of risk for the general population - 80 kilometers (radiological accidents).

		No Action	Decentralization			92/93 Planning Basis			Regionalization - A			Centralization			
Accident Description	Attribute	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c	Option 5a	Option 5b	Option 5c	Option 5d
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}
	Adjusted Annual Frequency	1.6×10^{-1}	3.3×10^{-1}	3.5×10^{-1}	1.6×10^{-1}	4.0×10^{-1}	4.0×10^{-1}	2.3×10^{-1}	4.4×10^{-1}	4.4×10^{-1}	2.8×10^{-1}	8.4×10^{-1}	8.4×10^{-1}	6.8×10^{-1}	1.7×10^{-1}
	Adjusted Point Estimate of Risk[b]	1.4×10^{-3}	2.8×10^{-3}	3.0×10^{-3}	1.4×10^{-3}	3.4×10^{-3}	3.4×10^{-3}	2.0×10^{-3}	3.7×10^{-3}	3.7×10^{-3}	2.4×10^{-3}	7.2×10^{-3}	7.2×10^{-3}	5.8×10^{-3}	1.4×10^{-3}
A2 - Processing release	Adjusted Health Effects[a]	(c)	(c)	(c)	2.6×10^{-4}	(c)	(c)	2.6×10^{-4}	(c)	(c)	2.6×10^{-4}	(c)	(c)	2.6×10^{-4}	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	2.7×10^{-1}	(c)	(c)	2.7×10^{-1}	(c)	(c)	2.7×10^{-1}	(c)	(c)	3.5×10^{0}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	7.0×10^{-5}	(c)	(c)	7.0×10^{-5}	(c)	(c)	7.0×10^{-5}	(c)	(c)	9.1×10^{-4}	(c)
A3 - Dry vault release	Adjusted Health Effects[a]	(c)	3.6×10^{-6}	(c)	3.6×10^{-6}	3.7×10^{-6}	(c)	3.7×10^{-6}	3.6×10^{-6}	(c)	3.6×10^{-6}	4.8×10^{-5}	(c)	4.8×10^{-5}	(c)
	Adjusted Annual Frequency	(c)	1.4×10^{-3}	(c)	1.4×10^{-3}	1.4×10^{-3}	(c)	1.4×10^{-3}	1.4×10^{-3}	(c)	1.4×10^{-3}	1.4×10^{-3}	(c)	1.4×10^{-3}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	5.0×10^{-9}	(c)	5.0×10^{-9}	5.0×10^{-9}	(c)	5.1×10^{-9}	5.0×10^{-9}	(c)	5.0×10^{-9}	6.7×10^{-8}	(c)	6.7×10^{-8}	(c)
A4 - Adjacent facility release	Adjusted Health Effects[a]	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}
	Adjusted Annual Frequency	2.4×10^{-3}	5.0×10^{-3}	5.3×10^{-3}	2.5×10^{-3}	5.9×10^{-3}	5.9×10^{-3}	3.4×10^{-3}	6.6×10^{-3}	6.6×10^{-3}	4.2×10^{-3}	1.3×10^{-2}	1.3×10^{-2}	1.0×10^{-2}	2.5×10^{-3}
	Adjusted Point Estimate of Risk[b]	6.0×10^{-5}	1.2×10^{-4}	1.3×10^{-4}	6.2×10^{-5}	1.5×10^{-4}	1.5×10^{-4}	8.5×10^{-5}	1.7×10^{-4}	1.7×10^{-4}	1.1×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	2.5×10^{-4}	6.2×10^{-5}

Table 5-29. (continued).

Accident Description	Attribute	No Action: Option 1	Decentralization: Option 2a	Option 2b	Option 2c	92/93 Planning Basis: Option 3a	Option 3b	Option 3c	Regionalization - A: Option 4a	Option 4b	Option 4c	Centralization: Option 5a	Option 5b	Option 5c	Option 5d
A5 - Criticality in water	Adjusted Health Effects[a]	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$
	Adjusted Annual Frequency	$3.1x10^{-3}$	$6.4x10^{-3}$	$6.8x10^{-3}$	$3.2x10^{-3}$	$7.7x10^{-3}$	$7.7x10^{-3}$	$4.4x10^{-3}$	$8.6x10^{-3}$	$8.6x10^{-3}$	$5.5x10^{-3}$	$1.6x10^{-2}$	$1.6x10^{-2}$	$1.3x10^{-2}$	$3.3x10^{-3}$
	Adjusted Point Estimate of Risk[b]	$1.4x10^{-5}$	$2.8x10^{-4}$	$3.0x10^{-4}$	$1.4x10^{-4}$	$3.4x10^{-4}$	$3.4x10^{-4}$	$1.9x10^{-4}$	$3.8x10^{-4}$	$3.8x10^{-4}$	$2.4x10^{-5}$	$7.0x10^{-4}$	$7.0x10^{-5}$	$5.7x10^{-4}$	$1.5x10^{-5}$
A6 - Criticality during processing	Adjusted Health Effects[a]	(c)	(c)	(c)	$4.3x10^{-3}$	(c)	(c)	$4.3x10^{-3}$	(c)	(c)	$4.3x10^{-3}$	(c)	(c)	$4.3x10^{-3}$	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	$1.5x10^{-4}$	(c)	(c)	$1.5x10^{-4}$	(c)	(c)	$1.4x10^{-4}$	(c)	(c)	$1.9x10^{-3}$	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	$6.5x10^{-7}$	(c)	(c)	$6.5x10^{-7}$	(c)	(c)	$6.0x10^{-7}$	(c)	(c)	$8.2x10^{-6}$	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[a]	$9.0x10^{-3}$	$9.4x10^{-3}$	$9.4x10^{-3}$	$9.4x10^{-3}$	$9.5x10^{-3}$	$9.5x10^{-3}$	$9.5x10^{-3}$	$9.3x10^{-3}$	$9.3x10^{-3}$	$9.3x10^{-3}$	$1.2x10^{-1}$	$1.2x10^{-1}$	$1.2x10^{-1}$	$1.2x10^{-1}$
	Adjusted Annual Frequency	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$
	Adjusted Point Estimate of Risk[b]	$1.8x10^{-6}$	$1.9x10^{-6}$	$1.9x10^{-6}$	$1.9x10^{-6}$	$1.9x10^{-6}$	$1.9x10^{-6}$	$1.9x10^{-6}$	$1.9x10^{-6}$	$1.9x10^{-6}$	$1.9x10^{-6}$	$2.4x10^{-5}$	$2.4x10^{-5}$	$2.4x10^{-5}$	$2.4x10^{-5}$
A8 - Internal spill/liquid discharge	Adjusted Health Effects[a]	$1.0x10^{-9}$	$1.0x10^{-9}$	$1.0x10^{-9}$	$1.0x10^{-9}$	$1.1x10^{-9}$	$1.1x10^{-9}$	$1.1x10^{-9}$	$1.0x10^{-9}$	$1.0x10^{-9}$	$1.0x10^{-9}$	$1.4x10^{-8}$	$1.4x10^{-8}$	$1.4x10^{-8}$	$1.4x10^{-8}$
	Adjusted Annual Frequency	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$
	Adjusted Point Estimate of Risk[b]	$1.1x10^{-10}$	$1.1x10^{-10}$	$1.1x10^{-10}$	$1.1x10^{-10}$	$1.2x10^{-10}$	$1.2x10^{-10}$	$1.2x10^{-10}$	$1.1x10^{-10}$	$1.1x10^{-10}$	$1.1x10^{-10}$	$1.5x10^{-9}$	$1.5x10^{-9}$	$1.5x10^{-9}$	$1.5x10^{-9}$

Table 5-29. (continued).

Accident Description	Attribute	Regionalization - B Option 4d	Option 4e	Option 4f	Option 4g
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	$8.5x10^{-3}$	$8.5x10^{-3}$	$8.5x10^{-3}$	$8.5x10^{-3}$
	Adjusted Annual Frequency	$4.1x10^{-1}$	$4.1x10^{-1}$	$2.5x10^{-1}$	$1.7x10^{-1}$
	Adjusted Point Estimate of Risk[b]	$3.5x10^{-3}$	$3.5x10^{-3}$	$2.1x10^{-3}$	$1.4x10^{-3}$
A2 - Processing Release	Adjusted Health Effects[a]	(c)	(c)	$2.6x10^{-4}$	(c)
	Adjusted Annual Frequency	(c)	(c)	$3.4x10^{-1}$	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	$8.8x10^{-5}$	(c)
A3 - Dry vault Release	Adjusted Health Effects[a]	$4.6x10^{-6}$	(c)	$4.6x10^{-6}$	(c)
	Adjusted Annual Frequency	$1.4x10^{-3}$	(c)	$1.4x10^{-3}$	(c)
	Adjusted Point Estimate of Risk[b]	$6.4x10^{-4}$	(c)	$6.4x10^{-4}$	(c)
A4 - Adjacent Facility Release	Adjusted Health Effects[a]	$2.5x10^{-2}$	$2.5x10^{-2}$	$2.5x10^{-2}$	$2.5x10^{-2}$
	Adjusted Annual Frequency	$6.2x10^{-3}$	$6.2x10^{-3}$	$3.7x10^{-3}$	$2.5x10^{-3}$
	Adjusted Point Estimate of Risk[b]	$1.6x10^{-4}$	$1.6x10^{-4}$	$9.2x10^{-5}$	$6.3x10^{-5}$

Table 5-29. (continued).

Accident Description	Attribute	Regionalization - B Option 4d	Option 4e	Option 4f	Option 4g
A5 - Criticality in water	Adjusted Health Effects[a]	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$	$4.4x10^{-3}$
	Adjusted Annual Frequency	$8.0x10^{-3}$	$8.0x10^{-3}$	$4.8x10^{-3}$	$3.3x10^{-3}$
	Adjusted Point Estimate of Risk[b]	$3.5x10^{-5}$	$3.5x10^{-5}$	$2.1x10^{-5}$	$1.4x10^{-5}$
A6 - Criticality during processing	Adjusted Health Effects[a]	(c)	(c)	$4.3x10^{-3}$	(c)
	Adjusted Annual Frequency	(c)	(c)	$1.8x10^{-4}$	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	$7.7x10^{-7}$	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[a]	$1.2x10^{-2}$	$1.2x10^{-2}$	$1.2x10^{-2}$	$1.2x10^{-2}$
	Adjusted Annual Frequency	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$	$2.0x10^{-4}$
	Adjusted Point Estimate of Risk[b]	$2.4x10^{-6}$	$2.4x10^{-6}$	$2.4x10^{-6}$	$2.4x10^{-6}$
A8 - Internal spill/liquid discharge	Adjusted Health Effects[a]	$1.3x10^{-9}$	$1.3x10^{-9}$	$1.3x10^{-9}$	$1.3x10^{-9}$
	Adjusted Annual Frequency	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$
	Adjusted Point Estimate of Risk[b]	$1.4x10^{-10}$	$1.4x10^{-10}$	$1.4x10^{-10}$	$1.4x10^{-10}$

a. Units for adjusted health effects are given in terms of potential fatal cancers.
b. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
c. The accident scenario is not included in the spectrum of potential accidents for this case.

unresolved DOE policy issues. For example, this cumulative impact assessment does not consider long-term reconfiguration issues. Table 5-30 presents a summary of cumulative impacts associated with the various spent fuel management alternatives.

5.16.1 Land Use

The land committed to spent nuclear fuel management activities at the SRS would lie, for the most part, within existing onsite industrial compounds or undeveloped onsite areas devoted to the continued mission of the Site. Under two of the alternatives - Regionalization by Location (at SRS) and Centralization (at SRS) - a new Expended Core Facility could be required to examine and characterize spent nuclear fuels from naval installations east of the Mississippi. Two locations have been proposed for the Expended Core Facility, one in the approximate center of the SRS and the other at the old Allied General Nuclear Services facility (or "Barnwell Nuclear Fuel Plant") that is located off Road G (and near SRS Barricade 4) just east of and adjacent to the Site.

Previously-undeveloped land committed to new spent nuclear fuel facilities (excluding the Expended Core Facility) would be limited to a maximum of approximately 100 acres (0.4 square kilometer). Depending on the location chosen, an additional 30 acres (0.1 square kilometer) could be required for a new Expended Core Facility. Thus, a maximum of 130 acres (0.5 square kilometer) could be converted from woodlands or old fields to industrial facilities and supporting infrastructure under the bounding options, Option 5a (Centralization - Dry Storage) and Option 5c (Centralization - Processing). Any site used for the support of spent nuclear fuel activities would be under government control. With the exception of the Barnwell Nuclear Fuel facility, which the Navy would purchase from Allied General Nuclear Services for an offsite Expended Core Facility, DOE would not require any additional land from the public domain for SRS spent nuclear fuel management facilities.

Ground was broken for the new Savannah River Ecology Laboratory Conference Center in May 1994. The new facility will occupy a 70-acre area, but only 5 to 10 acres will be cleared and graded for the new conference center, parking areas, and an access road. The remaining 60-65 acres will be managed as a nature study area and preserve. Thus, the Savannah River Ecology Laboratory Conference Center will require conversion of 5 to 10 acres of planted pines or pine/mixed hardwood (depending on the exact location of the building) to light-industrial/public use.

Table 5-30. Cumulative impacts associated with construction and operation of spent fuel alternatives at Savannah River Site.

ALTERNATIVE 1 - NO ACTION	
	Option 1 **Wet Storage**
Land Use	No new land committed to new use.
Socioeconomics	A maximum of 50 new jobs created annually during construction; no new jobs created during operation.
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.
Materials and Waste Management	High-Level: Current generation levels Transuranic: Current generation levels Low-Level: Current generation levels Mixed: Current generation levels Hazardous: Current generation levels Sanitary: Current generation levels

ALTERNATIVE 2 - DECENTRALIZATION			
	Option 2a Dry Storage	Option 2b Wet Storage	Option 2c Processing
Land Use	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.
Socioeconomics	Construction jobs: 600 peak Operation: No new jobs	Construction jobs: 600 peak Operation: No new jobs	Construction jobs: 550 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 4.4×10^{-4} rem.
Materials and Waste Management	High-Level: No change Transuranic: 6% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: No change Transuranic: 6% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: 475% increase Transuranic: 12% increase Low-Level: 100% increase Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]

Table 5-30. (continued).

ALTERNATIVE 3 - 1992/1993 PLANNING BASIS			
	Option 3a Dry Storage	Option 3b Wet Storage	Option 3c Processing
Land Use	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.
Socioeconomics	Construction jobs: 600 peak Operation: No new jobs	Construction jobs: 650 peak Operation: No new jobs	Construction jobs: 550 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 4.5×10^{-4} rem.
Materials and Waste Management	High-Level: No change Transuranic: 6% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: No change Transuranic: 6% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: 325% increase Transuranic: 12% increase Low-Level: 87.5% increase Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]

ALTERNATIVE 4 - REGIONALIZATION			
	Option 4a Dry Storage	Option 4b Wet Storage	Option 4c Processing
Land Use	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.
Socioeconomics	Construction jobs: 650 peak Operation: No new jobs	Construction jobs: 650 peak Operation: No new jobs	Construction jobs: 550 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 4.7×10^{-4} rem.

Table 5-30. (continued).

	Option 4a Dry Storage	Option 4b Wet Storage	Option 4c Processing
Materials and Waste Management	High-Level: No change Transuranic: No change Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: No change Transuranic: No change Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: 475% increase Transuranic: 6% increase Low-Level: 97.5% increase Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]

	Option 4d Dry Storage	Option 4e Wet Storage	Option 4f Processing
Land Use	Approximately 40 acres committed to new use.	Approximately 35 acres committed to new use.	Approximately 35 acres committed to new use.
Socioeconomics	Construction jobs: 910 peak Operation: No new jobs	Construction jobs: 910 peak Operation: No new jobs	Construction jobs: 860 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 4.7×10^{-4} rem.
Materials and Waste Management	High-Level: No change Transuranic: No change Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: No change Transuranic: No change Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: 475% increase Transuranic: 6% increase Low-Level: 97.5% increase Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]

	Option 4g Ship Out
Land Use	Less than one acre of land committed to new use.
Socioeconomics	Construction jobs: 200 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be (less than) $<9.0\times10^{-5}$ rem.

Table 5-30. (continued).

Materials and Waste Management	High-Level: Reduced volume of waste produced Transuranic: 6% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	

ALTERNATIVE 5 - CENTRALIZATION

	Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing
Land Use	100-130 acres of land committed to new use.	70-80 acres of land committed to new use.	100-130 acres of land committed to new use.
Socioeconomics	Construction: 2,550 peak Operation: No new jobs	Construction: 2,700 peak Operation: No new jobs	Construction: 2,550 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 4.7×10^{-4} rem.
Materials and Waste Management	High-Level: Reduced volume of waste produced Transuranic: Reduced volume of waste produced Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: 475% increase Transuranic: 18% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: 475% increase Transuranic: 18% increase Low-Level: 100% increase Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]

	Option 5d Ship Out
Land Use	Less than one acre of land committed to new use.
Socioeconomics	Construction: 200 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.
Materials and Waste Management	High-Level: Reduced volume of waste produced Transuranic: 6% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]

a. Not expected to change; no analysis conducted.
b. Not expected to change; based on projected employment levels at SRS.

Construction on the new Centralized Sanitary Wastewater Treatment Facility is scheduled to begin in 1994 and should be completed in 1995. This new facility will be built approximately 1 mile south of F-Area on Burma Road. Building the central facility will require clearing approximately 6 acres of planted pines. An 18 mile trunkline/collection system will also be required, using existing transmission line and steam line rights-of-way to the extent possible. This trunkline will be located in the northwest quadrant of the SRS, and will connect the new Centralized Sanitary Wastewater Treatment Facility to A-Area, F-/H-Areas, and C-Area.

Depending on the spent nuclear fuel management alternative chosen, a total of 150 acres of SRS land could be cleared and converted to facilities and infrastructure as a result of spent nuclear fuel management (including an Expended Core Facility), construction of the Savannah River Ecology Laboratory Conference Center, and completion of the Centralized Sanitary Wastewater Treatment Facility. This represents less than 0.1 percent of the undeveloped land on the SRS, and will have minimal cumulative impact on long-term land use locally and regionally.

5.16.2 Socioeconomics

There would be minimal cumulative impacts on the socioeconomic resources of the SRS region from any spent fuel management alternative. The greatest change in employment would occur under the Centralization Alternative, which would include construction and operation of an Expended Core Facility at SRS. Construction of an Expended Core Facility would require an estimated 850 additional employees in the peak year (1999), while operation of the facility would add a maximum of approximately 500 full-time jobs. DOE anticipates that overall employment on the Site will decline during the first 5 years of the spent fuel management period and will stabilize thereafter as the SRS mission changes. Workers who might otherwise lose their jobs could be employed by SRS in spent fuel program activities. Therefore, DOE expects little or no direct increase in employment due to the program. The Site would fill any new jobs from the existing regional labor force.

5.16.3 Air Quality

Table 5-31 compares the cumulative emissions of nonradioactive pollutants from the SRS, including those from the proposed spent nuclear fuel alternatives, to the pertinent regulatory standards. The values provided are the maximum concentrations that would occur at ground level at the Site boundary. Not all maximum concentrations would occur at the same location.

Table 5-31. Total maximum ground-level concentrations (μg/cubic meter) of criteria and toxic air pollutants at SRS boundary resulting from normal operations and spent nuclear fuel management alternatives.[a,b]

Emissions	Averaging Time	Alternatives 1 through 4 Option a Dry Storage	Option b Wet Storage	Option c Processing
Criteria Pollutants				
NO_x	Annual	4 (4%)	4 (4%)	15 (15%)
SO_x	Annual	10 (12%)	10 (12%)	10 (12%)
	24-hours	185.0 (50%)	185.0 (50%)	185.4 (50%)
	3-hours	634 (49%)	634 (49%)	637 (49%)
PM_{10}	Annual	3 (6%)	3 (6%)	3 (6%)
	24-hours	56.0 (37%)	56.0 (37%)	56.4 (37%)
TSP	Annual	11 (17%)	11 (17%)	11 (17%)
Ozone (as VOC)	1-hour	N/A[d]	N/A[d]	N/A[d]
Gaseous fluoride (as HF)	1-month	0.03 (4%)	0.03 (4%)	0.05 (6%)
	1-week	0.15 (9%)	0.15 (9%)	0.25 (16%)
	24-hours	0.31 (11%)	0.31 (11%)	0.51 (18%)
	12-hours	0.62 (17%)	0.62 (17%)	1.02 (28%)
Lead	Annual	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)
CO	8-hours	23.1 (0.2%)	23.1 (0.2%)	27.3 (0.3%)
	1-hour	181 (0.4%)	181 (0.4%)	212 (0.5%)
Toxic Pollutants				
Nitric acid	24-hours	6.7 (5%)	6.7 (5%)	7.7 (6%)
1,1,1-Trichloroethane	24-hours	22 (0.2%)	22 (0.02%)	22 (0.2%)
Benzene	24-hours	31 (21%)	31 (21%)	31 (21%)
Ethanolamine	24-hours	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)
Ethylbenzene	24-hours	0.12 (<0.1%)	0.12 (<0.1%)	0.12 (<0.1%)
Ethylene glycol	24-hours	0.08 (<0.1%)	0.08 (<0.1%)	0.08 (<0.1%)
Formaldehyde	24-hours	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)
Glycol ethers	24-hours	<0.01 N/A	<0.01 N/A	<0.01 N/A
Hexachloronaphthalene	24-hours	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)
Hexane	24-hours	0.07 (<0.1%)	0.07 (<0.1%)	0.11 (<0.1%)
Manganese	24-hours	0.10 (0.4%)	0.10 (0.4%)	0.10 (0.4%)
Methanol	24-hours	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)
Methyl ethyl ketone	24-hours	0.99 (<0.1%)	0.99 (<0.1%)	0.99 (<0.1%)
Methyl isobutyl ketone	24-hours	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)
Methylene chloride	24-hours	1.8 (0.3%)	1.8 (0.3%)	1.82 (0.4%)
Napthalene	24-hours	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)
Phenol	24-hours	0.03 (<0.1%)	0.03 (<0.1%)	0.03 (<0.1%)

Table 5-31. (continued).

Emissions	Averaging Time	Alternatives 1 through 4: Option a Dry Storage	Option b Wet Storage	Option c Processing
Phosphorus	24-hours	<0.001 (<0.2%)	<0.001 (<0.2%)	<0.001 (<0.2%)
Sodium hydroxide	24-hours	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)
Toluene	24-hours	1.6 (8%)	1.6 (8%)	2.0 (10%)
Trichloroethene	24-hours	1.0 (0.3%)	1.0 (0.3%)	1.0 (0.3%)
Vinyl acetate	24-hours	0.02 (<0.1%)	0.02 (<0.1%)	0.02 (<0.1%)
Xylene	24-hours	3.81 (<0.1%)	3.81 (<0.1%)	3.85 (<0.1%)

Emissions	Averaging Time	Alternative 5 - Centralization: Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing	Option 5d Ship Out
Criteria Pollutants					
NO_x	Annual	4 (4%)	4 (4%)	15.1 (15%)	4 (4%)
SO_x	Annual	10 (12%)	10 (12%)	10 (12%)	10 (12%)
	24-hours	185.0 (50%)	185.0 (50%)	185.5 (52%)	185.0 (50%)
	3-hours	634.5 (49%)	634.5 (49%)	637.5 (49%)	634 (49%)
PM_{10}	Annual	3 (6%)	3 (6%)	3 (6%)	3 (6%)
	24-hours	56.0 (37%)	56.0 (37%)	56.4 (38%)	56.0 (37%)
TSP	Annual	11 (17%)	11 (17%)	11 (17%)	11 (17%)
Ozone (as VOC)	1-hour	N/A[d]	N/A[d]	N/A[d]	N/A[d]
Gaseous fluoride (as HF)	1-month	0.03 (4%)	0.03 (4%)	0.05 (6%)	0.03 (4%)
	1-week	0.15 (9%)	0.15 (9%)	0.25 (16%)	0.15 (9%)
	24-hours	0.31 (11%)	0.31 (11%)	0.41 (14%)	0.31 (11%)
	12-hours	0.62 (17%)	0.62 (17%)	1.02 (28%)	0.62 (17%)
Lead	Annual	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)
CO	8-hours	24 (0.2%)	24 (0.2%)	28.1 (0.3%)	23.1 (0.2%)
	1-hour	187 (0.5%)	187 (0.5%)	217 (0.5%)	181 (0.4%)
Toxic Pollutants					
Nitric acid	24-hours	6.7 (5%)	6.7 (5%)	7.7 (6%)	6.7 (5%)
1,1,1-Trichloroethane	24-hours	22 (0.2%)	22 (0.02%)	22 (0.2%)	22 (0.2%)
Benzene	24-hours	31 (21%)	31 (21%)	31 (21%)	31 (21%)
Ethanolamine	24-hours	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)
Ethylbenzene	24-hours	0.12 (<0.1%)	0.12 (<0.1%)	0.12 (<0.1%)	0.12 (<0.1%)
Ethylene glycol	24-hours	0.08 (<0.1%)	0.08 (<0.1%)	0.08 (<0.1%)	0.08 (<0.1%)
Formaldehyde	24-hours	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)
Glycol ethers	24-hours	<0.01 (N/A)	<0.01 (N/A)	<0.01 (N/A)	<0.01 (N/A)
Hexachloronaphthalene	24-hours	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)

Table 5-31. (continued).

Emissions	Averaging Time	Alternative 5 - Centralization			
		Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing	Option 5d Ship Out
Hexane	24-hours	0.07 (<0.1%)	0.07 (<0.1%)	0.11 (<0.1%)	0.07 (<0.1%)
Manganese	24-hours	0.10 (0.4%)	0.10 (0.4%)	0.10 (0.4%)	0.10 (0.4%)
Methanol	24-hours	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)
Methyl ethyl ketone	24-hours	0.99 (<0.1%)	0.99 (<0.1%)	0.99 (<0.1%)	0.99 (<0.1%)
Methyl isobutyl ketone	24-hours	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)
Methylene chloride	24-hours	1.8 (0.3%)	1.8 (0.3%)	1.82 (0.4%)	1.8 (0.3%)
Napthalene	24-hours	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)
Phenol	24-hours	0.03 (<0.1%)	0.03 (<0.1%)	0.03 (<0.1%)	0.03 (<0.1%)
Phosphorus	24-hours	<0.001 (<0.2%)	<0.001 (<0.2%)	<0.001 (0.2%)	<0.001 (<0.2%)
Sodium hydroxide	24-hours	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)
Toluene	24-hours	1.6 (8%)	1.6 (8%)	2.0 (10%)	1.6 (8%)
Trichloroethene	24-hours	1.0 (0.3%)	1.0 (0.3%)	1.0 (0.3%)	1.0 (0.3%)
Vinyl acetate	24-hours	0.02 (<0.1%)	0.02 (<0.1%)	0.02 (<0.1%)	0.02 (<0.1%)
Xylene	24-hours	3.81 (<0.1%)	3.81 (<0.1%)	3.85 (<0.1%)	3.81 (<0.1%)

a. Source: WSRC (1994a).
b. Numbers in parentheses indicate the percentage of the regulatory standard that each concentration represents.
c. No standard for this chemical.
d. Measurement data currently unavailable.

The data demonstrate that, even with the emissions from the spent nuclear fuel management activities, releases of toxic air pollutants from the SRS would be only a small fraction of the regulatory standards. Therefore, DOE anticipates no cumulative impact.

The releases of some criteria air pollutants by SRS operations would approach regulatory standards. Site sulfur dioxide emissions would reach about 50 percent of both the 24-hour and 3-hour limits under all alternatives. In addition, the emissions of particulates less than 10 microns (PM_{10}) would approach a concentration equal to about 38 percent of the standard. However, the contribution to both these pollutants concentrations made by spent nuclear fuel-related activities would be small, as explained in Section 5.7.

The SRS evaluated the cumulative impact of airborne radioactive releases in terms of cumulative dose to a maximally exposed individual at the Site boundary. Table 5-32 lists the results of this

Table 5-32. Annual cumulative health effects to workers and offsite population due to SRS radioactive releases during incident-free operations.

	Worker				Offsite Population			
	Average Individual		Total Collective		Maximally Exposed Individual		Total Collective	
	Dose[a]	Fatal Cancer[b]	Dose[c]	Fatal Cancers[d]	Dose[a]	Fatal Cancer[b]	Dose[c]	Fatal Cancers[d]
Alternative 1 - No Action								
Option 1 Wet Storage	3.2×10^{-1}	1.3×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Alternative 2 - Decentralization								
Option 2a Dry Storage	3.0×10^{-1}	1.2×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 2b Wet Storage	3.2×10^{-1}	1.3×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 2c Processing	3.6×10^{-1}	1.5×10^{-4}	1.6×10^{2}	6.5×10^{-2}	4.4×10^{-4}	2.2×10^{-7}	2.6×10^{1}	1.3×10^{-2}
Alternative 3 - 1992/1993 Planning Basis								
Option 3a Dry Storage	3.0×10^{-1}	1.2×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 3b Wet Storage	3.2×10^{-1}	1.3×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 3c Processing	3.7×10^{-1}	1.5×10^{-4}	1.6×10^{2}	6.6×10^{-2}	4.5×10^{-4}	2.2×10^{-7}	2.6×10^{1}	1.3×10^{-2}
Alternative 4 - Regionalization								
Option 4a Dry Storage	3.0×10^{-1}	1.2×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 4b Wet Storage	3.2×10^{-1}	1.3×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 4c Processing	3.7×10^{-1}	1.5×10^{-4}	1.7×10^{2}	6.8×10^{-2}	4.7×10^{-4}	2.3×10^{-7}	2.7×10^{1}	1.4×10^{-2}
Option 4d Dry Storage	3.2×10^{-1}	1.3×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 4e Wet Storage	3.5×10^{-1}	1.4×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{8}	8.9×10^{0}	4.4×10^{3}
Option 4f Processing	4.0×10^{-1}	1.6×10^{-4}	1.7×10^{2}	6.8×10^{-2}	4.7×10^{-4}	2.3×10^{-7}	2.6×10^{1}	1.3×10^{-2}
Option 4g Ship Out	$<3.2 \times 10^{-1}$	$<1.3 \times 10^{-4}$	$<9.4 \times 10^{1}$	$<3.7 \times 10^{-2}$	$<9.0 \times 10^{-5}$	$<4.5 \times 10^{-8}$	$<8.9 \times 10^{0}$	$<4.4 \times 10^{-3}$

Table 5-32. (continued).

	Worker				Offsite Population			
	Average Individual		Total Collective		Maximally Exposed Individual		Total Collective	
	Dose[a]	Fatal Cancers[b]	Dose[c]	Fatal Cancers[d]	Dose[a]	Fatal Cancers[b]	Dose[c]	Fatal Cancers[d]
Alternative 5 - Centralization								
Option 5a Dry Storage	1.3	$5.3x10^{-4}$	$9.6x10^{1}$	$3.8x10^{-2}$	$9.0x10^{-5}$	$4.5x10^{-8}$	$8.9x10^{0}$	$4.4x10^{-3}$
Option 5b Wet Storage	1.6	$6.4x10^{-4}$	$9.6x10^{1}$	$3.8x10^{-2}$	$9.0x10^{-5}$	$4.5x10^{-8}$	$8.9x10^{0}$	$4.4x10^{-3}$
Option 5c Processing	1.6	$6.6x10^{-4}$	$1.7x10^{2}$	$6.9x10^{-2}$	$4.7x10^{-4}$	$2.3x10^{-7}$	$2.7x10^{1}$	$1.4x10^{-2}$
Option 5d Ship Out	$<3.2x10^{-1}$	$<1.3x10^{-4}$	$<9.4x10^{1}$	$<3.7x10^{-2}$	$<9.0x10^{-5}$	$<4.5x10^{-8}$	$<8.9x10^{0}$	$<4.4x10^{-3}$

a. Dose in rem.
b. Probability of fatal cancer.
c. Dose in person-rem.
d. Incidence of excess fatal cancers.

analysis. The highest dose would be $4.7x10^{-1}$ millirem, which would occur under the processing options of Alternatives 4 and 5. This dose is below the regulatory standard (CFR 1994) of 10 millirem.

Airborne emissions from the two-unit Vogtle Electric Generating Plant (approximately 10 miles southwest of the center of the SRS near Waynesboro, Georgia) were reported to have delivered an MEI total body dose of 1.14×10^{-3} millirem during 1992 (Georgia Power Company 1993). Since the SRS and Plant Vogtle are essentially proximal to the same 80 kilometer population, the ratio of SRS population and MEI doses was used as an estimator of the population dose due to Plant Vogtle emissions. Using this approach, the population dose attributable to Vogtle was estimated to have been about 8.3×10^{-2} person-rem in 1992. Adding (1) the population dose from Plant Vogtle, (2) the total collective offsite population dose from all SRS activities in 1992 (both air and water source terms), and (3) the highest projected collective dose from spent nuclear fuel management activities (Options 4c and 5c) yields a total cumulative dose of 27.083 person-rem from all SRS sources and Plant Vogtle, which is only 0.3 percent higher than the dose from SRS alone. Note that the doses in Table 5-32 ("Total Collective Dose, Offsite Population") represent the sum of (2) and (3) above.

5.16.4 Water Resources

Approximately 82.1 million gallons per year of Savannah River water would be required for the two most water-intensive options, Option 4f (Regionalization at SRS - Processing) and Option 5c (Centralization - Processing). Because either of these options would probably require construction of an Expended Core Facility, this facility's projected surface water usage of 2.5 million gallons per year was factored into the cumulative impacts analysis. Thus, the two options with the highest surface water usage, both of which would require as much as 84.6 million gallons, represent approximately 0.4 percent of the current (baseline) SRS surface water usage of 20 billion gallons per year (see Table 5-8).

Operational impacts to surface water quality under any of the spent nuclear fuel management options examined would be minimal. Existing SRS treatment facilities could accommodate all new spent nuclear fuel-related domestic and process wastewater streams. Expected wastewater flows would be well within the design capacities of existing (or planned upgrades of) Site treatment systems. Sanitary wastewater from new spent nuclear fuel facilities would be routed to the new Centralized Sanitary Wastewater Treatment Facility. Liquid radioactive wastes would presumably be sent to the F-/H-Area Effluent Treatment Facility. Treated nonradioactive liquid releases from the new spent nuclear fuel facilities would likely be discharged to Upper Three Runs Creek or Fourmile Branch.

Water quality in the Savannah River downstream of the SRS is adequate to good, with most parameters analyzed showing values below state and Federal Maximum Contaminant Levels or DOE Derived Concentration Guides. Iron, present in soils in the region, is the only constituent of surface waters that routinely exceeds MCLs. Spent nuclear fuel management activities are not expected to result in higher concentrations of iron downstream of the SRS. As noted earlier, in Section 5.16, construction on the new Centralized Sanitary Wastewater Treatment Facility is scheduled to begin in 1994 and should be completed in 1995. The new Centralized Sanitary Wastewater Treatment Facility will replace 14 aging sanitary wastewater facilities with a single state-of-the-art facility which will treat sanitary wastes by an extended aeration-activated sludge process. Chlorine will not be used to treat sanitary wastes in the new facility. Use of non-chemical ultraviolet light disinfection systems will eliminate the use and handling of 32,000 gallons of sodium hypochlorite and 59,000 gallons of sodium sulfite per year. Eliminating these chemicals will essentially eliminate the potential for toxic chemical releases from the wastewater treatment process.

Operation of the new Centralized Sanitary Wastewater Treatment Facility and closure of the old A-, B-, S-Area, and Naval Fuel sanitary wastewater facilities would also eliminate wastewater discharges to Upper Three Runs Creek, the stream on the SRS least degraded by past operations. Treated effluent from the new Centralized Sanitary Wastewater Treatment Facility will discharge to Fourmile Branch. Overall stream quality in Fourmile Branch is expected to improve because the effluent from the new facility will be cleaner than the effluent from the old package plants in C-, F-, and H-Areas that presently discharge to Fourmile Branch. As a result, the cumulative effect of the new spent nuclear fuel management facilities (any alternative considered) and new Centralized Sanitary Wastewater Treatment Facility will probably be a net improvement in water quality in two SRS streams, Upper Three Runs Creek and Fourmile Branch, and may result in better water quality downstream in the Savannah River as well.

Sanitary wastewater from the new Consolidated Incineration Facility will be routed to the new Centralized Sanitary Wastewater Treatment Facility; there will be no direct process wastewater drains to the environment. Liquid wastes will be collected in storage tanks and periodically trucked to a permitted hazardous/mixed waste treatment and disposal facility. Sanitary wastes from the new Savannah River Ecology Laboratory Conference Center will be piped to a septic tank-drain field system and would not impact surface water in the area.

Sanitary wastes produced during construction of the Expended Core Facility would be treated through the use of portable chemical toilets or through an existing wastewater treatment facility. Depending on the location chosen by DOE and the Navy for the new Expended Core Facility, sanitary wastes from operation of the ECF would either be treated in an existing wastewater treatment facility (most likely the new Centralized Sanitary Wastewater Facility) or a new treatment facility designed to handle the facility's wastewater capacity. No process wastes from operation of the Expended Core Facility will be discharged to the environment.

5.16.5 Occupational and Public Health and Safety

Table 5-32 summarizes the cumulative health effects of incident-free SRS operations, including those projected for the spent nuclear fuel alternatives. The table lists potential cancer fatalities for workers and the public due to radiological exposures to airborne and waterborne releases from the Site. In addition, the table provides the (airborne) dose to the hypothetical maximally exposed individual in the offsite population. The evaluation used 1992 as the baseline year for normal operations, because it is the last year for which the SRS has complete information. DOE believes that this year gives a realistic depiction of current operational releases of radionuclides. The assessment added the estimated releases from each spent fuel alternative to this baseline to determine the cumulative impacts listed in Table 5-32.

5.16.6 Waste Management

The analysis of cumulative impacts of SRS waste management activities takes as its starting point the assumption that waste generation under the No Action Alternative represents the baseline condition for the entire Savannah River Site. Waste generation levels associated with the other proposed spent nuclear fuel management alternatives (see Table 5-19) thus represent positive and negative deviations from this baseline. Cumulative effects of the proposed spent nuclear fuel alternatives on the volume of low-level waste, transuranic waste, and high-level waste produced under each of the proposed alternatives are presented in Table 5-30.

In addition to baseline waste generation and wastes generated by spent nuclear fuel management activities, environmental restoration and cleanup activities are expected to become an increasingly important part of the DOE mission at the SRS in the future. These remediation activities are expected to produce large quantities of radioactive, hazardous, and mixed wastes. It is estimated that approximately 22,000 cubic meters (28,754 cubic yards) of low-level waste, 366,000 cubic meters (478,362 cubic yards) of hazardous waste, 82,000 cubic meters (107,174 cubic yards) of mixed wastes, and 900 cubic meters (1,176 cubic yards) of transuranic wastes would be produced by environmental restoration activities at the SRS over the 1995-2024 period (DOE 1995). Decontamination and decommissioning activities are expected to generate approximately 109,000 cubic meters (142,463 cubic yards) of low-level waste, 32,000 cubic meters (41,824 cubic yards) of hazardous waste, 95,000 cubic meters (124,165 cubic yards) of mixed wastes, and 4,000 cubic meters (5,228 cubic yards) of transuranic wastes over the same 30-year period (DOE 1995). High-level radioactive waste would not be generated by environmental restoration or decontamination and decommissioning activities.

5.17 Unavoidable Adverse Environmental Impacts

The construction and operation of facilities related to any of the five alternatives at the Savannah River Site (SRS) would result in some adverse impacts to the environment. Changes in project design and other measures could eliminate, avoid, or reduce most of these to minimal levels. The following paragraphs identify adverse impacts that mitigation could not reduce to minimal levels or avoid altogether.

The generation of some fugitive dust during construction would be unavoidable, but would be controlled by water and dust suppressants. This would occur under Alternatives 2 to 5, but greatest generation of dust would occur under Alternative 5 (excluding the offsite shipping option). Similarly, construction activities would result in some minor, yet unavoidable, noise impacts from heavy equipment, generators, and vehicles.

The maximum loss of habitat would involve the conversion of 70 to 100 acres (0.28 to 0.4 square kilometer) of managed pine forest to industrial land use; this would occur under Alternative 5 if DOE moved all spent nuclear fuel to the SRS.

The amount of radioactivity that normal operation of the spent nuclear fuel facilities would release under four of the five alternatives (Alternatives 1 to 4) would be a small fraction of the 1992 operational releases at the SRS and would be well below applicable regulatory standards.

For the alternative having the most impact (Alternative 5 - Centralization), DOE has calculated that the maximum probability for latent fatal cancer for the maximally exposed member of the public would be about 3 times higher than that calculated for 1992 at the SRS. For latent fatal cancer incidence in the offsite population, this comparison indicates an increase of about 2 times, but the number of cancers calculated is less than one.

The only socioeconomic impacts of the proposed spent nuclear fuel management facilities would be temporary increases in employment and expenditures in the region of influence during the construction phase. These would be unavoidable beneficial impacts.

5.18 Relationship Between Short-Term Use of the Environment and the Maintenance and Enhancement of Long-Term Productivity

Implementation of any of the proposed alternatives would result in some short-term resource demands (e.g., fuel, construction materials, and labor) and would, under certain alternatives (notably the Centralization Alternative), reduce the natural productivity of a relatively small tract of land (less than .07 percent of total SRS area) currently committed to timber production. Depending upon the precise location selected for facility development, a small amount of marginal-to-good wildlife habitat (see Sections 4.9 and 5.9) would also be lost when the area is cleared, graded, and committed to facilities and supporting infrastructure. However, these short-term resource losses and land-use restrictions provide a basis for improved productivity and utility over the long term at the SRS because consolidating all spent nuclear fuel at a few onsite locations would free for other uses those locations presently committed to spent fuel management. On a national scale, the interim management plan described in this EIS would have the same impact of making locations throughout the DOE complex available for other long-term uses.

5.19 Irreversible and Irretrievable Commitments of Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of facilities related to the spent nuclear fuel alternatives would involve materials that could not be recovered or recycled or that would be consumed or reduced to unrecoverable forms. The construction and operation of spent nuclear fuel facilities at the SRS would consume irretrievable amounts of electrical energy, fuel, concrete, sand, gravel, and miscellaneous chemicals. Other resources used in construction would probably not be recoverable. These would include finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery. Construction and operation of facilities for spent nuclear fuel management would also require the withdrawal of water from surface- and groundwater sources, but most of this water would return to onsite surface streams or the Savannah River after use and treatment.

The Centralization alternative (Option 5c - Processing) would consume the greatest amount of electricity of any of the alternatives, about 110,400 megawatt-hours. The Processing option (excluding Option 4c, Regionalization by fuel type) would have the highest requirements for coal to produce steam, approximately 2,580 metric tons (2,843 tons) annually. The Centralization alternative (except Option 5d where all spent fuel would be shipped off the site) would involve the greatest irretrievable consumption of other resources, such as construction materials, chemicals, gases, and operating supplies. However, this demand would not constitute a permanent drain on local resources or involve any material that is in short supply in the region.

5.20 Potential Mitigation Measures

This section summarizes measures that DOE could use to avoid or reduce impacts to the environment caused by spent nuclear fuel management activities at the SRS. DOE would determine the extent to which any mitigation would be necessary and the selection of which measures would be implemented during a detailed site-specific NEPA review tiered from this Programmatic EIS. Consequently, the following sections in this chapter address impact avoidance and mitigation in general terms and describe typical measures that the SRS could implement. In addition, the analyses described in this appendix indicate that the environmental consequences of spent fuel management would be minimal in most environmental media.

5.20.1 Pollution Prevention

DOE is committed to comply with Executive Order 12856, "Federal Compliance with Right-to-Know Laws and Pollution Prevention Requirements"; Executive Order 12780, "Federal Acquisition, Recycling and Waste Prevention"; and applicable DOE Orders and Guidance Documents in planning and implementing pollution prevention at the SRS. The pollution prevention program at the Site was initiated in 1990 as a waste minimization program. Currently, the program consists of four major initiatives: solid waste minimization; source reduction and recycling of wastewater discharges; source reduction of air emissions; and potential procurement of products manufactured from recycled materials. Since 1991, the waste of all types generated at the SRS has decreased, with greatest reductions in hazardous and mixed wastes. These reductions are attributable primarily to material substitutions.

All spent fuel management activities at the SRS would be subject to the Site pollution prevention program. Implementation of the program plan would minimize the amount of waste generated by these activities.

5.20.2 Socioeconomics

Spent nuclear fuel activities would have minimal impact on the socioeconomic environment in the region of influence because most employees would be drawn from the existing site workforce. The minor impacts of in-migrating construction workers could be minimized by DOE possibly informing local communities and county planning agencies as to scheduling of construction activities.

5.20.3 Cultural Resources

A Programmatic Memorandum of Agreement (SRARP 1989) between the DOE Savannah River Operations Office, the South Carolina State Historic Preservation Office, and the Advisory Council on Historic Preservation, ratified on August 24, 1990, is the instrument for the management of cultural resources at the SRS. DOE uses this memorandum to identify cultural resources and develop mitigation plans for affected resources in consultation with the State Historic Preservation Officer. DOE would comply with the terms of the memorandum for all measures needed to support spent nuclear fuel management at the Site. For example, DOE would survey sites prior to disturbance and could reduce impacts to any potentially-significant cultural resources discovered through avoidance or removal. Any artifacts discovered would be protected from further disturbance and the elements until removed.

DOE conducted an investigation of Native American concerns over religious rights in the Central Savannah River Valley in conjunction with studies in 1991 related to a New Production Reactor. During this study, three Native American groups expressed concern over sites and items of religious significance on the SRS (see Section 4.4.2). DOE has included these organizations on its environmental mailing list, solicits their comments on NEPA actions of the Site, and sends them documents about SRS environmental activities, including those related to these SNF management considerations. These Native American groups would be consulted on any actions that may follow subsequent site-specific environmental reviews.

5.20.4 Geology

DOE expects that there would be no impacts to geologic resources at the SRS under any alternative evaluated in this EIS. Potential soil erosion in areas of ground disturbance would be minimized through sound engineering practices such as implementing controls for stormwater runoff (e.g., sediment barriers), slope stability (e.g., rip-rap placement), and wind erosion (e.g., covering soil stockpiles). Re-landscaping would minimize soil loss after construction was completed. These measures would be included in a site-specific Storm Water Pollution Prevention Plan that the SRS would prepare prior to initiating any construction.

5.20.5 Air Resources

DOE would meet applicable standards and permit limits for all radiological and non-radiological releases to the atmosphere. In addition, the SRS would follow the DOE policy of maintaining radiological emissions to levels "as low as reasonably achievable" (ALARA). ALARA is an approach to radiation protection to control or manage exposures (both individual and collective) and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit. ALARA is not a dose limit, but rather a process that has as its objectives the attainment of dose levels as far below the applicable limits as practicable.

5.20.6 Water Resources

DOE would minimize the potential for adverse impacts on surface water during construction through the implementation of a stormwater pollution prevention plan that details controls for erosion and sedimentation. The plan would also establish measures for prevention of spills of fuel and chemicals and for rapid containment and cleanup.

DOE could minimize water usage during both construction and operation of facilities by instituting water conservation measures such as instructing workers in water conservation (e.g., turn off hoses when not in use), installing flow restrictors, and using self-closing hose nozzles.

5.20.7 Ecological Resources

DOE does not anticipate that any of the spent fuel alternatives would impact any wetlands on the Site. In any case, DOE and SRS policy is to achieve "no net loss" of wetlands. Pursuant to this goal, DOE has issued a guidance document, *Information for Mitigation of Wetlands Impacts at the Savannah River Site* (DOE 1992), for project planners that puts forth a practical approach to wetlands protection that begins with avoidance of impacts (if possible), moves to minimization of impacts (if avoidance is impossible), and requires compensatory measures (wetlands restoration, creation, or acquisition) in the event that impacts cannot be avoided.

The analysis in this EIS indicates that there are no threatened and endangered species or sensitive habitats in the areas considered as representative of potential sites for spent nuclear fuel activities at the SRS. However, DOE would perform site-specific predevelopment surveys to ensure that development of new facilities would not impact any of these biological resources.

5.20.8 Noise

DOE anticipates that noise impacts both on and off the Site would be minimal. DOE does not foresee noise impacts from spent nuclear fuel management that would warrant mitigation measures beyond those consistent with good construction, engineering, operations, and management practices.

5.20.9 Traffic and Transportation

DOE has a system of onsite buses operating at the SRS. The Site would evaluate the need for upgrades or changes in service that might be required for the spent nuclear fuel management activities and would make changes, as necessary.

DOE would manage changes in traffic volume or patterns during construction through such measures as designating routes for construction vehicles, providing workers with safety reminders, and upgrading onsite police traffic patrols, if necessary.

5.20.10 Occupational and Public Health and Safety

The DOE program for maintaining radiological emissions to levels "as low as reasonably achievable" (ALARA) described in Section 5.20.5 above will minimize any impacts to workers and the public due to atmospheric releases. Likewise, the Site Pollution Prevention Plan and emergency preparedness measures will enhance safety both on and off the Site.

5.20.11 Utilities and Support Services

The utilities and support services at the SRS are sufficient to meet the requirements of any of the alternatives for the spent fuel management at the Site. Impacts on these services would be minimal. No mitigation measures would be required.

5.20.12 Accidents

The SRS has in place emergency action plans that would be activated in the case of an accident. These plans contain both onsite provisions (e.g., evacuation plans, response teams, medical and fire response, training and drills, communications equipment) and offsite arrangements (e.g., response plans for medical and fire agencies, coordination with local and state agencies, communication plans). The

SRS plans would be updated to include any new facilities or activities related to spent nuclear fuel management that would involve the Site. The execution of the plans in response to an accident would mitigate adverse effects both on the Site and in the surrounding areas.

ATTACHMENT A: ACCIDENT ANALYSIS

A.1 Accident Evaluation Methodologies and Assumptions

The potential for facility accidents and the magnitude of their consequences is an important factor in the evaluation of the spent nuclear fuel alternatives addressed in this EIS. There are two health risk issues:

- Would accidents at any of the Savannah River Site (SRS) facilities that the U.S. Department of Energy (DOE) could build for spent nuclear fuel management activities pose unacceptable health risks to workers or the general public?

- Could alternative locations or facilities for the spent nuclear fuel alternatives provide smaller public or worker health risks? Smaller risks could arise from such factors as greater isolation of the facility from the public, a reduced frequency of such external accident initiators as seismic events or aircraft crashes, reduced inventory, and process differences.

Guidance for the implementation of Council on Environmental Quality (CEQ) regulations (CFR 1986), as amended (51 FR 15625), requires the evaluation of impacts that would have a low probability of occurrence but high consequences if they did occur; this EIS, therefore, addresses facility accidents to the extent feasible.

A.1.1 Radiological Accident Evaluation Methodology

The alternatives considered in this EIS provide an opportunity to incorporate new features and technology in new facilities, processes, and operations that would minimize the possibility of undue risk to the health and safety of plant workers and the public. Modifications and upgrades would mitigate accident consequences from existing facilities or reduce the likelihood of occurrence.

Under normal circumstances, DOE would develop accident scenarios and calculate accident consequences using safety analyses, mitigation features, and design details on proposed facility designs. However, the preliminary design information for the proposed facilities that is available during the preparation of this EIS does not contain sufficient detail to permit quantitative safety analyses. Therefore, for each spent nuclear fuel alternative, DOE has evaluated the existing and proposed facilities for the type of radiological accidents it has determined to be reasonably foreseeable.

The radiological accident types fell into four categories: (1) fuel damage, (2) material releases, (3) nuclear criticalities, and (4) liquid spills or discharges. For each accident type, DOE determined reference accidents by examining DOE-approved safety analysis reports (SARs) and other appropriate documentation (e.g., previous EISs). In addition, DOE considered accidents from adjacent facilities for their possible impacts related to spent nuclear fuel. DOE extracted the overall frequency for each reference accident from the appropriate source, rather than attempting to calculate individual frequencies for all possible initiators; that is, DOE did not use the specific probability of a certain magnitude earthquake to determine the frequency of a criticality or spill, given the occurrence of the earthquake. If multiple initiators could lead to one of the reference accidents, or the combined frequency of the initiators could lead to one of the reference accidents, DOE used the combined frequency of the initiators, generally providing conservative results. For example, the Receiving Basin for Offsite Fuel has a number of potential release initiators that could result in an uncontrolled criticality, as listed in Table A-1. As listed, a number of incidents, all of which have their own assigned frequencies, can contribute to the initiation of an uncontrolled criticality.

Table A-1. Potential release initiators at the Receiving Basin for Offsite Fuel.

Natural Phenomena	External Events	Operations Induced Events	Criticality
Temperature Extreme	Aircraft Crash	Fuel Cutting	Fuel Bundling Error
Snow	Helicopter Crash	Spill at Hose Rack	Cask Loading Error
Rain	Surface Vehicle Crash	Fuel Rupture in Storage	Fuel Identification Problem
Lightning		Fire and Explosion	Fuel Movement Error
Tornado		Fuel Near Basin Surface	Dropped Fuel
Earthquake		Spills and Leaks	Crane or Hoist Collapse
Meteorite Impact		Resin Regeneration Facility Waste to Cell	Cask Immersion Error

This evaluation results in qualitative comparisons for proposed facilities based on the assumption that the facility function is similar to one already analyzed. In addition, an identical set of initiators is not considered in each safety analysis report for existing SRS facilities because these reports were prepared over several years in accordance with requirements in effect at the time. Section A.2

includes a comparison of the similarities of possible facilities to an existing facility, the basis for the selection of reference accidents, and several tables containing data to support a comparison of point estimates of risk.

The qualitative comparison supports the National Environmental Policy Act (NEPA) process, in that the decisionmaker can assess the relative risk from each alternative at SRS and other sites.

A.1.1.1 Notable Accident Initiators. While there are many different types of accident initiators of various frequencies that could lead to an accident, three notable initiators - criticalities, earthquakes, and aircraft crashes - require additional discussion due to the public's perception of the importance of these initiators and the public's familiarity with these types of initiators.

Because there has never been an uncontrolled criticality accident at the SRS, DOE must use historic experience related to the initiators to estimate the frequency for a criticality incident in the Receiving Basin for Offsite Fuel. Storage basins for spent nuclear fuel have excellent safety histories. From 1945 through 1980, there were 40 known criticality accidents worldwide, none of which occurred in a fuel storage facility. From 1975 to 1980, there were, conservatively, 160 reactors with storage basins in operation around the world, and no criticality incidents occurred. Therefore, DOE assumes that the upper frequency limit for a criticality event is 3.1×10^{-3} per year (Du Pont 1983). This figure is applicable to the extent that the storage basins and the operations performed in them are similar to those of the Receiving Basin for Offsite Fuel. However, the frequency for a processing criticality event was determined through a detailed fault tree analysis, as referenced in the safety analysis report, to be an overall calculated limit of 1.4×10^{-4} per year. This value accounts for the implementation of new administrative controls or equipment.

The SRS is in an area that has a relatively low seismic frequency. Based on three centuries of recorded seismic activity, an earthquake with a Richter magnitude greater than 6.0, which corresponds to a Modified Mercalli Intensity Scale (MMI) of VII, would not be likely at the SRS. The design-basis earthquake for the SRS is a MMI VIII event with a corresponding horizontal peak ground acceleration of $0.2g$. Based on current technology, as applied in various probabilistic evaluations of the seismic hazard in the SRS region, the $0.2g$ peak ground acceleration can be associated with a 2×10^{-4} annual probability of exceedance (5,000-year return period). There are four scenarios for the

Receiving Basin for Offsite Fuel to which an earthquake of intensity MMI VIII or greater might contribute:

- Deformation of the storage racks leading to a criticality incident.

- Derailment of the 100-ton (91-metric-ton) crane into the storage basin with the deformation of the storage rack leading to criticality.

- Damage to the basin walls leading to the release of contaminated basin water to the subsoil.

- Rupture of a waste tank or pipe in the Resin Regeneration Facility leading to the release of contaminated liquids.

An aircraft crash into a spent nuclear fuel facility is of concern because it could result in a radioactive release of materials from the stored spent nuclear fuel. Appendix D contains an aircraft crash probability analysis based on the examination of large civilian and military aircraft crossing the airspace within a 10-mile (16-kilometer) radius of the SRS. It does not include the crash probability of general aviation aircraft because aircraft of this type generally do not possess sufficient mass or attain sufficiently high velocities to produce a serious radiological threat in the event that they crashed into an area containing spent nuclear fuel. The analysis did not evaluate crash probabilities with a likelihood of occurrence of less than 10^{-7} per year because they would not significantly contribute to the risk. This was the case for spent nuclear fuel facilities located at the SRS.

A.1.1.2 Use of DOE-Approved Safety Documents. The NEPA guidance issued by the DOE Office of NEPA Oversight, dated May 1993, recommends that accident impact analyses "reference Safety Assessments and Safety Analysis Reports, if available." This guidance was the primary basis used to develop the approach used in the accident analysis section of this EIS. This Appendix uses several relevant safety analysis reports as well as a previously published EIS. Safety analysis reports are the primary source of information on reasonably foreseeable accidents with the potential to cause a release of hazardous materials. These reports are required for all reactors and nuclear materials facilities with operations that potentially pose a significant hazard to onsite personnel, offsite populations, or the environment. The referenced safety analysis reports and EIS approval/draft submittal dates encompass a range from 1983 to 1993. The 1983 safety analysis report was supplemented by a 1993 addendum; the next oldest safety analysis report was approved in 1988.

A.1.2 Chemical Hazard Evaluation Methodology

This analysis reviewed the appropriate safety analyses to assess the degree to which they addressed chemical accidents. It found that each of the safety analyses addressed chemical hazards in a qualitative manner. To provide a quantitative discussion of chemical hazards, the analysis evaluated a separate risk assessment (WSRC 1993c) for the storage risk of offsite research reactor fuel in the Receiving Basin for Offsite Fuel to determine a bounding chemical accident. The analysis determined chemical inventories (see Section A.3) for the existing spent nuclear fuel facilities at the SRS using the "Savannah River Site Tier Two Emergency and Hazardous Chemical Inventory Report" (WSRC 1994a) to determine the facilities total chemical inventory. This chemical inventory was further screened using the EPA's "List of Lists" (EPA 1990).

A.1.3 SRS Emergency Plan

The SRS emergency plan (WSRC 1993b) defines appropriate response measures for the management of emergencies (e.g., accidents) involving the Site. It incorporates into one document a description of the entire process designed to respond to and mitigate the consequences of an accident. Emergencies that could cause activation of all or portions of this plan include:

- Events (operational, transportation, etc.) with the potential to cause releases above allowable limits of hazardous materials.

- Events such as ftres, explosions, tornadoes, hurricanes, earthquakes, dam failures, etc., that affect or could affect safety systems designed to protect site and offsite populations and the environment.

- Events such as bomb threats, hostage situations, etc., that reduce the security posture of the Site.

- Events created by proximity to other facilities, such as the Vogtle Electric Generating Plant, a commercial nuclear powerplant located across the Savannah River from the Site.

For radiological emergencies, protective actions in this plan are designed to keep onsite and offsite exposures As Low As Reasonably Achievable (ALARA). This is accomplished by minimizing time spent in the vicinity of the hazard, keeping as far from the hazard as possible, and taking advantage of available shielding. Protective actions that could be used on the Site in the event of an emergency include remaining indoors, sheltering, evacuation, and relocation. For events that cause an actual or projected radiological release, appropriate protective actions for on- and offsite populations have been determined based on trigger points called Protective Action Guides (PAGs).

A.1.4 General Assumptions

This assessment applied the following key assumptions to examine existing accident analyses and to relate these analyses to the spent nuclear fuel alternatives.

- When a referenced accident scenario is used for a possible new facility, DOE would build the new facility close to an existing referenced facility performing a similar function, resulting in consequences and health effects similar to the existing facilities analyzed. The exception could be the proposed Expended Core Facility which Appendix D analyzes separately.

- For existing facilities to be modified, portions of the facility to be decommissioned, or new facilities to be added, potential accident initiators resulting from construction and nearby activities would be bounded by the referenced accident scenarios.

- Type 2 High Enriched Uranium fuel, the dominant type currently in storage or process at the SRS, would provide a reference source term for other fuel types (i.e., Mark-22 fuel).

- Spent nuclear fuel acceptance criteria would specify that all fuel must be capable of indefinite suspension in air with no melting.

- The total frequency of an event (e.g., criticality) could be used to determine point estimates of risk, regardless of the type or specific frequencies of the individual contributing initiators.

- Adjustment (scaling) factors could be applied to reflect a best engineering judgment in terms of relative risk between the various alternatives.

- The point estimate of risk for a given accident scenario would be representative in that it could, for the purposes of this programmatic EIS, represent a similar accident scenario at new facilities that perform similar functions.

- Reference accidents would be attributed to a facility based on its function (e.g., fuel canning or dry material storage) regardless of whether the facility currently exists, is undergoing design, or is in the conceptual design phase.

- Possible new facilities would be designed to pose no greater risk to the workers and public than existing facilities with similar functions.

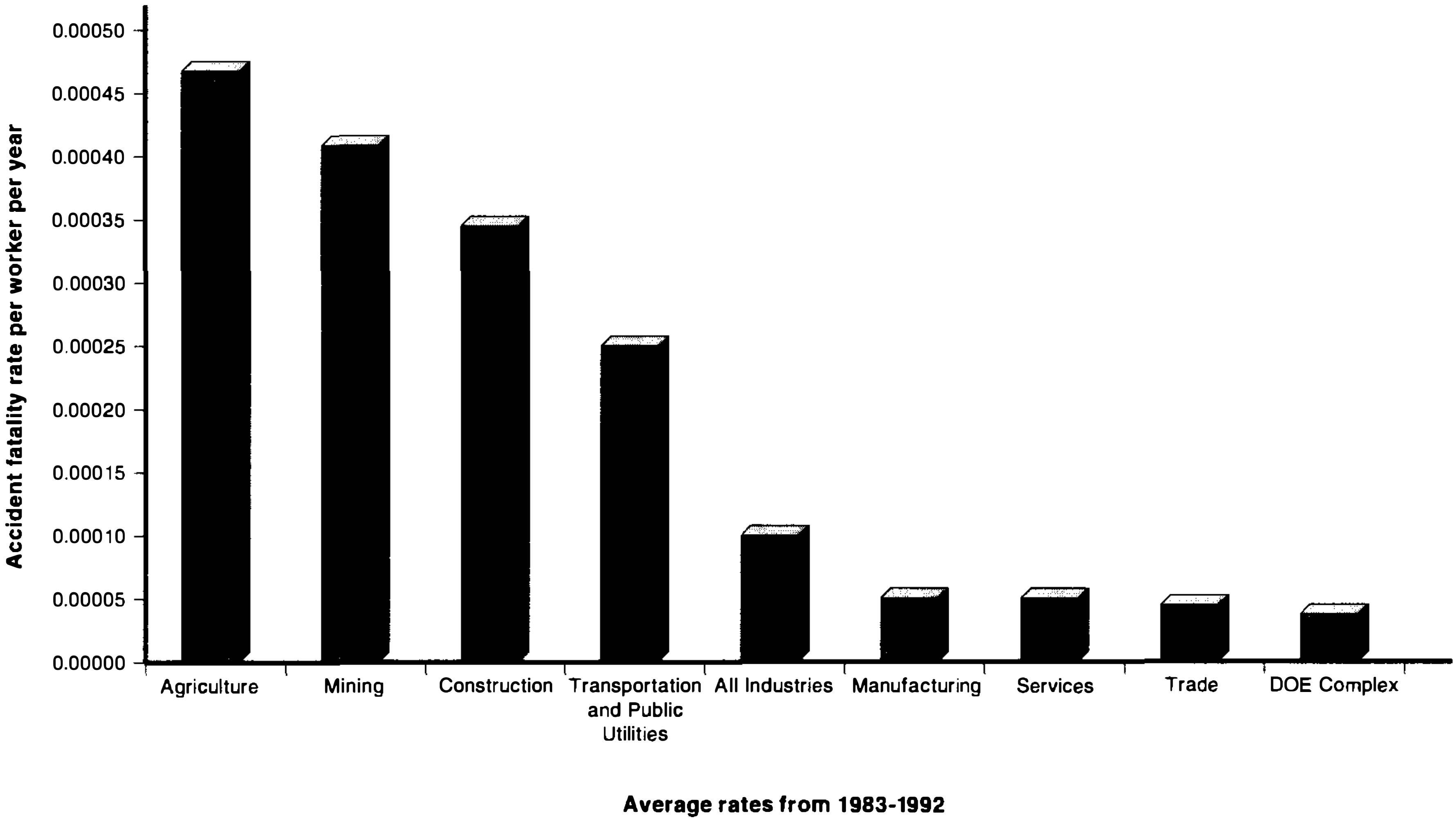
This evaluation takes no credit for the upgraded design requirements for the proposed facilities. Such facilities should have improved reliability or mitigative features and, therefore, would reduce the aggregate frequency of accidents. Therefore, the application of values from existing safety analysis reports would provide conservative results. In addition, the evaluation makes no attempt to discriminate among similar existing facilities that might have slightly different frequencies of occurrence or source terms (i.e., an FB-Line event frequency was applied to HB-Line and other processing facilities).

For most accidents, the evaluation did not quantify consequences for workers. The safety analysis reports from which information was extracted for the reference accidents were written before the issuance of DOE Order 5480.23 (DOE 1992); previous applicable Orders did not require the inclusion of worker doses. The historic record indicates that DOE facilities have an enviable safety record. Figure A-1 compares the rate of worker fatalities in the DOE complex (DOE 1993) to national average rates compiled by the National Safety Council for various industry groups (NSC 1993). Because the DOE worker accident fatality rate compares favorably to rates from such industry groups as agriculture and construction and is slightly less than trade and services group rates, the absence of quantitative data regarding accident impacts to radiological workers should not impede the decisionmaking process. The discussion presented in Volume 1 adequately addresses the impacts for close-in workers (i.e., those directly involved in the activity or near the accident source) at the SRS.

A.1.4.1 Receptor Group Assumptions. To ensure comparative results, the evaluation assessed the measures of impacts among four receptor groups:

- Worker. An individual located 100 meters (328 feet) in the worst sector of a facility location where the release occurs.

- Colocated Worker. An individual located 640 meters (2,100 feet) in the worst sector of a facility location where the release occurs.



Source: NSC (1993); DOE (1993)

PK54-5

Figure A-1. Comparison of fatality rates among workers in various industry groups.

- Maximally Exposed Offsite Individual (MEI). A hypothetical resident located at the nearest Site boundary from the facility location where the release occurs.

- Offsite Population to 80 Kilometers. The collective sum of individuals located within an 80-kilometer (50-mile) radius of the SRS.

As noted above, the worker is 100 meters (328 feet) from the facility where the accident occurs. This is because information quantifying accident impacts (i.e., dose and health effects) to workers at less than 100 meters from an accidental release of radionuclides is unavailable. For each of the accident scenarios considered in Appendix C of this EIS, there is some risk of worker injury or death at distances closer than 100 meters. Furthermore, the safety analyses from which this evaluation extracted information for the accident scenarios often did not include any discussions on worker impacts as a result of potential accidents. DOE Orders published before DOE 5480.23 (DOE 1992) did not require the inclusion of worker doses. However, Section A.2.6.2 includes a qualitative discussion regarding accident impacts for the worker at less than 100 meters (328 feet) for each of the radiological accident scenarios.

A.1.4.2 Code Assumptions. DOE's application of the AXAIR and AXAIR89Q (a validated version) dose estimation models is acceptable for projecting health effects from accidents at SRS and comparing the results to results from other similar codes (RSAC-5 and GENII) used at other sites. AXAIR is a Gaussian model based on the methodology outlined in NRC Regulatory Guide 1.145 (NRC 1983). AXAIR contains a meteorological data file specific to SRS that provides conservative calculated doses for the radiological consequences of atmospheric releases. AXAIR and AXAIR89Q include the following specific functions:

- Performs both environmental transport and radiation dosimetry calculations

- Bases environmental transfer models on NRC Reg Guide 1.145 guidelines

- Includes exposure pathways for inhalation of radionuclides and gamma radiation from the radioactive plume

- Calculates gamma shine doses using a non-uniform Gaussian model

- Uses worst sector and 99.5-percentile meteorology

Doses calculated with this code should bound the radiological consequences for atmospheric releases postulated.

A.1.4.3 Criticality Assumptions. An estimate of the consequences of a criticality incident requires an estimate of the number of fissions that might occur. While U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.34 specifies 1×10^{19} fissions as the upper tenth of incidence experience, the SRS analyses are based on mean values, to the extent possible, for all incidents. Criticality incidents have produced from 10^{14} to 4×10^{19} fissions with a mean of 2×10^{18} fissions for incidents involving fissile solutions and a mean of 5×10^{17} fissions for incidents involving solids. As a consequence, two accident scenarios (Table A-2) address criticality - the wet pool criticality scenario and the processing criticality scenario. For the wet pool criticality scenario, the mean value for solid systems (5×10^{17}) is assumed to apply to the source term used to determine the accident consequences, while the processing criticality scenario assumes that the mean value for a solution (2×10^{18}) was applied to the source term to determine accident consequences.

A.2 Radiological Accident Scenarios

A.2.1 Selection of Reference Accidents

To support the examination of both existing and proposed facilities, this evaluation considered a spectrum of potential accident types. To develop a meaningful spectrum of potential accidents, the evaluation posed the following question:

> "What could be done to spent nuclear fuel that would result in a radiological consequence to the receptor groups?"

In determining the answer to this question, the following four general types of events emerged: (1) fuel damage, (2) material releases, (3) criticalities, and (4) liquid spills or discharges. A review of applicable safety analysis reports for the SRS facilities that the spent nuclear fuel alternatives would be likely to affect generated more than 20 accidents involving the transport, receipt, processing, and storage of spent nuclear fuel. A consolidation and subsequent "binning" of these accidents for each accident type reflects an appropriate range of case-specific reference accidents.

Table A-2. Reference radiological accidents considered for spent nuclear fuel activities.

	Name and Reference	Reference for Source Term/Dose	Comparative Likelihood/Frequency
A1.	Fuel Assembly Breach Reference Accident: RBOF fuel cutting	Tables 1-3 DPSTSA-200-10-3, Addendum 1	1.6×10^{-1} per year
A2.	Material Release (Processing) Reference Accident: F-Canyon Uncontrolled Reaction	Meehan 1995	2.6×10^{-1} per year
A3.	Material Release (Dry Vault) Reference Accident: PSF release	Table 5-9 DPSTSA-200-10-19	1.4×10^{-3} per year
A4.	Material Release (Adjacent Facility) Reference Accident: Release of Waste Tank Activity to Cell	Tables 1-3 DPSTSA-200-10-3, Addendum 1	2.4×10^{-3} per year
A5.	Criticality in Water Reference Accident: RBOF criticality	Tables 1-3 DPSTSA-200-10-3, Addendum 1	3.1×10^{-3} per year
A6.	Criticality During Processing Reference Accident: FB-Line	WSRC-RP-93-1102	1.4×10^{-4} per year
A7.	Spill/Liquid Discharge (External) Reference Accident: Direct discharge of water from K-Reactor disassembly basin	Figure 3 Meehan 1994	2.0×10^{-4} per year
A8.	Spill/Liquid Discharge (Internal) Reference Accident: RBOF hose rack spill	Tables 1-3 DPSTSA-200-10-3, Addendum 1	1.1×10^{-1} per year

The fuel damage event (type 1 accident) considered was physical damage or breaching of a fuel assembly. Three material (type 2 accidents) releases were considered; they represent releases that could occur during processing from medium energetic events, those that could occur during dry storage of special nuclear materials, and those that could occur from an adjacent facility. Criticality (type 3 accidents) can have different dose impacts and can occur with different frequencies, depending on the physical or chemical characteristics of the material and the surroundings. Two criticality events - in water and during processing - represent these accident scenarios. The evaluation considered a dry criticality accident scenario bounded by the wet pool criticality in terms of frequency and bounded by the processing criticality accident in terms of number of fissions assumed. Two liquid discharges and spills (type 4 accidents) were considered - discharges of pool or basin water assumed to contain tritium, cesium, and other radioactive constituents from the fuel in the pool (external spill), and spills of slightly contaminated liquids inside a facility during fuel handling, spraying, or cask unloading (internal spill).

These eight typical accidents form the set of accidents for the selection of a reference accident. Each type has been assigned an alphanumeric designator, which is listed below and used throughout this document:

- Type 1 - Fuel damage

  A1 - Fuel assembly breach

- Type 2 - Material releases

  A2 - Processing release
  A3 - Dry vault release
  A4 - Adjacent facility release

- Type 3 - Criticalities

  A5 - Criticality in water
  A6 - Criticality during processing

- Type 4 - Liquid discharges and spills

  A7 - External spill/liquid discharge
  A8 - Internal spill/liquid discharge

A second review of the safety analyses and the original list of accidents confirmed that each specific accident considered in DOE-approved safety analyses could be represented or bounded by one of the eight "generic" accidents (i.e., a fire could result in material release or an earthquake could result in criticality or liquid release). The use of this approach with documented total frequencies avoids the need for unique identification of all initiating precursor events or their specific probabilities.

A.2.1.1 Externally Initiated Accidents. The accident analysis section of this EIS considered accident scenarios from external events or adjacent facilities and their potential impacts on direct spent nuclear fuel activities and facilities. Three significant sources of externally induced accident mechanisms were identified as potentially applicable to these facilities and activities: aircraft crashes, adjacent fires, and adjacent explosions. As discussed above, an aircraft crash scenario is not a reasonably foreseeable event within the probability scope of this EIS. For the most part, a fire or explosion in a facility adjacent to the spent nuclear fuel facilities described in Figure 3-2 would not have a significant impact on spent nuclear fuel facilities. However, the screening process determined that a fire and explosion in the Resin Regeneration Facility, located immediately adjacent to the Receiving Basin for Offsite Fuel, could result in the airborne release to the shielded cell and should be included for completeness.

A.2.1.2 Nearby Industrial or Military Facility Accidents. Within a 40-kilometer (25-mile) radius of the SRS, there are approximately 120 industrial facilities with 25 or more employees (DOE 1990). Four of these facilities are within a 16-kilometer (10-mile) radius of the SRS. Other than those on the SRS, the only major storage facilities within a 40-kilometer radius are the facilities at Chem-Nuclear Systems, Inc., Vogtle Electric Generating Station, and a cluster of natural gas storage tanks near Beech Island. The facilities within a 16-kilometer radius of the SRS boundary are still at least 10 kilometers (6 miles) from the nearest spent nuclear fuel facility, and thus present negligible risk to spent nuclear fuel activities.

A.2.1.3 Common Cause Accident. DOE considered accident scenarios based on a common cause accident during the screening process. A severe seismic event was the only common-cause initiator identified with the potential to simultaneously impact multiple spent nuclear fuel management facilities at the SRS. A design basis earthquake, which has an estimated acceleration of 0.2*g* and an annual frequency of 2.0×10^{-4} per year (or one occurrence every 5,000 years), could potentially impact multiple facilities within a single facility area, resulting in the simultaneous release of radioactive and/or toxic materials from these facilities to the environment. It is also considered possible, although probably less likely, than an earthquake of the same magnitude could damage facilities in more than one facility area (e.g., F- and H-Areas; K-, L-, and P-Reactor Disassembly Basins), resulting in simultaneous releases to the environment.

A semi-quantitative evaluation of the cumulative impacts resulting from multiple releases within an area caused by a severe seismic event was performed as part of the accident selection process described in Section A.2.1. A review of the safety analysis reports for the H-Canyon, HB-Line, and Receiving Basin for Offsite Fuels was performed to determine the consequences and risks presented individually by each facility following a design basis earthquake. The risks presented in each safety analysis report were then summed to approximate the risk that would be expected if all of these releases occurred simultaneously from a single seismic initiator. The sum of these risks was compared to the risks of the other accident scenarios presented within the EIS and were found to be bounded by those accidents. A similar evaluation was performed for the spent nuclear fuel-related facilities in the F-Area, and the same conclusion was reached. For the reactor disassembly basins, multiplying the risk from a severe earthquake calculated for the K-Reactor Disassembly basin by three could be considered as the outermost bounding estimate for the three reactor disassembly basins (K-, L-, and P-Reactor Disassembly Basins). This is considered an unrealistic estimate of the cumulative risk because of the extremely conservative assumptions that were made in performing the K-Reactor Disassembly Basin analysis (Meehan 1994). However, even if the risk is increased by a factor of three, it is still considered to be bounded by other accidents already presented within the EIS. Therefore, consistent with the accident methodology described in Section A.2.1, no further analysis of this type of scenario was required. The SRS does maintain emergency plans that would provide protective actions and mitigate consequences that could occur during a common cause accident scenario.

A.2.1.4 Accidents Resulting from Terrorism. DOE considered accident scenarios based on a terrorist attack or an act of sabotage during the screening process and concluded that any accident resulting from such initiators would be bounded by or similar to the accident scenarios already considered.

A.2.2 Reference Accident Descriptions

DOE established a reference accident for each of the eight generic or typical accidents. The following paragraphs outline the basis for selection of each reference accident by scenario. A reference accident was included if it is analyzed in an SRS safety analysis report that has been approved by the DOE or submitted to DOE for approval as part of the safety basis authorizing operation of a facility, and if the facility is to be utilized as, or is similar in function to, one of the facilities included in the five alternatives and their subordinate cases. For example, the analysis assumed that the Receiving Basin for Offsite Fuel was representative of any spent nuclear fuel wet storage pool. If an accident could occur in any pool, the analysis selected a reference scenario from the Receiving Basin for Offsite Fuel Safety Analysis Report as the reference accident, as listed in Table A-2. The following paragraphs provide the basis for each selection.

- **A1. Fuel Assembly Breach** - Physical damage to an assembly could occur from dropping, objects falling onto the assembly, or cutting into the fuel part of an assembly. The Receiving Basin for Offsite Fuel Safety Analysis Report (WSRC 1993a) Addendum contains a current analysis of a "fuel cutting accident." The inert, non-uranium-containing extremities of some spent nuclear fuel elements are cut off (cropped) in the repackaging basin before the bundling of the elements. The spent nuclear fuel could be inadvertently cut, causing a release of airborne or high water activity to the work area. Because of the metallic nature of SRS fuel, only a very small fraction of the gases generated in an assembly would be released to the basin water in an accident. Consistent with the safety analysis report, fuel cooled for 90 days is used in the source term for this accident. With foreign research reactor spent nuclear fuel elements, the release of fission product gases would be less than with the Mark-22 fuel assemblies previously considered. The physics of the release of gases from research reactor fuel is similar to SRS fuel because the fuel is constructed in a similar manner. Spent nuclear fuels that could release more fission gases than a Mark-22 fuel assembly would require an Unreviewed Safety Question analysis before the SRS could accept them in the Receiving Basin for Offsite Fuel. Air monitors in this area would warn personnel in the event of an airborne release. The fuel cutting operation involves only one fuel element at a time. This is representative for all cutting and dropping accidents because cracking the cladding would release less than cutting into the fuel itself.

A2. Material Release (Processing) - The primary activities associated with processing spent nuclear fuel include dissolving the fuel in acid in the F- or H-Area Canyon, separating the radioactive and fissile isotopes, and forming those isotopes into a solid material, either metal or powder. Because of the large volumes of liquid radioactive solution generated during the dissolution process, uncontrolled reactions in the Canyons are the most rapid means of losing control of the material and inadvertently releasing potentially significant quantities of material to the environment. The most common uncontrolled reactions, and those considered in this scenario, include eructations, foaming, boilover, and gassing while dissolving spent fuel. These types of uncontrolled reactions are typically caused by chemical addition errors, procedural errors, or equipment failure. Although uncontrolled reactions can also include deflagrations and explosions (caused by excess hydrogen generation due to radiolytic decay and the presence of an ignition source), these types of events are much less common, and because of their lower frequency, typically present a lower risk to workers and members of the public. In developing this scenario, it was assumed that the uncontrolled reaction causes a large release of material within the Canyon building to the Canyon sumps which results in a greater than normal release of radioactive material through the ventilation system and Canyon exhaust stack. In addition, it was assumed that the uncontrolled reaction occurred in the F-Canyon facility since the exposures resulting from an inadvertent release of plutonium isotopes are expected to bound potential inadvertent releases of uranium isotopes from uncontrolled reactions in the H-Canyon facility.

- **A3. Material Release (Dry Vault)** - Accident types A1 and A2 cover material releases from fuel handling and processing. In addition, DOE considered a reference accident for vault-type storage. The Plutonium Storage Facility (PSF) Safety Analysis Report (Du Pont 1989) analyzed three medium energetic events (shipping container failure, criticality, and impact-type events) and an earthquake. As discussed above, medium energetic events are accidents that result in release of material from the primary container and have sufficient energy to penetrate the secondary confinement barriers for a short period of time. That report contains a total frequency of these four initiating events and provides one release value. Because the SRS has no long-term spent nuclear fuel dry storage facilities, this evaluation assumes that the Plutonium Storage Facility vault is representative of dry storage facilities, as are the activities and precursor events. A material release from any medium energetic event in the Plutonium Storage Facility was selected as the reference accident for nonprocessing material releases.

- **A4. Material Release (Adjacent Facility)** - For completeness, DOE considered a reference accident from a facility immediately adjacent to the Receiving Basin for Offsite Fuel (WSRC 1993a). This scenario includes a fire and explosion at the Resin Regeneration Facility in waste tank EP 38 during which the coolant of a received cask, when discharged to the waste tank, results in a flammable or explosive concentration of vapors in the tank. Rupture of the tank by an explosion could release airborne activity to the shielded cell if the accident occurred during one of the projected 150 times per year when regeneration of the portable columns takes place. While a fire and explosion have not occurred in waste tank EP 38, one fire and pressure surge did occur when a shipping cask was being vented. The spent nuclear fuel remained intact and radionuclides were not released. The incident has been attributed to the ignition of a mixture of hydrogen, oxygen, and air emanating from the cask and created by reaction of hot aluminum fuel with water left in the cask by the shipper.

- **A5. Criticality in Water** - This scenario assumes that a wet pool storage facility is the most likely to have a criticality in water. The Receiving Basin for Offsite Fuel provides the capability for underwater receipt, handling, and storage of spent nuclear fuel. Primary radiation shielding is provided by the water covering the spent nuclear fuel. A safety analysis report determined frequency and results from many initiating events that could lead to criticality. The following activities could ultimately lead to a criticality incident: Fuel Bundling, Cask Loading, Fuel Identification and Manifest Problems, Fuel Movement, Dropped Fuel, Fuel Near Basin, Cask Immersion, and Cranes and Hoist. These events are representative for any wet storage pool.

- **A6. Criticality During Processing** - As noted in the discussion for accident type A2, FB-Line events are representative for SRS processing facilities. The analysis considered the total of the frequencies for criticality initiators for all processing stages, which would, therefore, be conservative because not all processing stages would necessarily be involved in a new facility and not all stages would necessarily occur simultaneously.

- **A7. Spill/Liquid Discharge (External)** - The reference accident selected for this type of event is the direct discharge of water (i.e., 3.4 million gallons) from the K-Reactor disassembly basin to the Savannah River and the exposure of fuel and targets in the basin to air. Analyses performed by the DOE while developing the EIS for the Interim Management of Nuclear Materials at the SRS demonstrate that this scenario could be initiated by a severe earthquake and would result in bounding airborne exposures (from exposed fuel) and liquid exposures (contaminated drinking water) to the general public. The selection of the direct-discharge event is conservative for existing or possible new facilities constructed in the F- or H-Areas because no free-flowing surface streams would be near a discharge point. The use of the source term from the reactor disassembly basin is considered to be conservative for the spent nuclear fuel storage pools since its inventory consists primarily of the fuel types with the largest source terms available for release (i.e., Mark-22 assemblies). Although the disassembly basin has water circulating systems to control radioactivity, chemistry, clarity, and temperature, these processes are less efficient than those used in the Receiving Basin for Offsite Fuel, resulting in higher concentrations of tritium, cesium, and other contaminants available for release.

- **A8. Spill/Liquid Discharge (Internal)** - DOE considered a second reference accident for contaminated liquids spills or discharges to ensure the appropriate onsite impacts. The discharge discussed for accident type A7 would be external to the building and would have no measurable worker impact component because the reference accident occurred outside the facility. The Receiving Basin for Offsite Fuel hose rack spill was selected as the reference accident because it is representative of small, unplanned, but relatively frequent spills in a storage facility and could impact the worker. Minor releases of contaminated water could occur at the hose rack platform during the handling of portable deionizers for the reactor areas.

A.2.3 Source Term and Frequency Determinations

Table A-2 lists source term references from existing documents approved by DOE or submitted by Westinghouse Savannah River Company to DOE for approval for each selected reference accident. The same references nominally prescribed the frequency of accidents or initiating events. If it was not directly available, the frequency was derived from information already contained in the appropriate safety analysis report or EIS (e.g., if only a risk estimate and a dose were listed, the frequency was derived by dividing the risk by the dose). These frequencies fall into ranges associated with abnormal events (more frequent than 1×10^{-3} per year), design-basis accidents (1×10^{-3} per year to 1×10^{-6} per year), or beyond-design-basis accidents (less than 1×10^{-6} per year to 10^{-7} per year).

This document does not analyze beyond-design-basis accidents or accidents with frequencies of less than 1.0×10^{-6} explicitly because the accident analysis source material (DOE-approved safety analysis reports) considers these accidents to be incredible events. Beyond-design-basis accidents, such as an airplane crash-induced criticality, have no different consequences (i.e., number of fissions) than the criticality estimated to occur with a frequency of 3.1×10^{-3} per year. Because of the use of aggregate frequencies in some cases, the contribution to overall risk from 1.0×10^{-7} per year events is negligible, and the higher frequency initiators dominate the point estimate of risk. Some initiating or precursor event frequencies from the safety analysis reports are at 10^{-7} per year or lower; thus, these reports in fact consider events beyond the 10^{-6} frequencies.

Frequencies for reference accidents were determined as follows:

- **A1. Fuel Assembly Breach** - The frequency for this reference accident was obtained from DPSTSA-200-10-3, *Receiving Basin for Offsite Fuel (RBOF)*, Addendum 1, Tables 1-5, which lists the frequency as 1.6×10^{-1} per year (WSRC 1993a).

- **A2. Material Release (Processing)** - The frequency for this reference accident was obtained from DPSTSA-200-10-4, *Safety Analysis - 200 Area, Savannah River Plant, F-Canyon Operations*, Addendum 2, "Accident Analysis," Revision 1, Table A.5.5-7A, which lists the frequency for an uncontrolled chemical reaction (the bounding processing accident) as 2.6×10^{-1} per year (Meehan 1995).

- **A3. Material Release (Dry Vault)** - The frequency for this reference accident was obtained from DPSTSA-200-10-19, *Final Safety Analysis Report - 200 Area, Savannah River Site Separations Area Operations, Building 221F, B-Line, Plutonium Storage Facility*, July 1989, Table 5-9, which lists the frequency as 1.4×10^{-3} per year (Du Pont 1989).

- **A4. Material Release (Adjacent Facility)** - The frequency for this reference accident was obtained from DPSTSA-200-10-3, *Receiving Basin for Offsite Fuel (RBOF)*, Addendum 1, Tables 1-5, which lists the frequency as 2.4×10^{-3} per year (WSRC 1993a).

- **A5. Criticality in Water** - The frequency for this reference accident was obtained from DPSTSA-200-10-3, *Receiving Basin for Offsite Fuel (RBOF)*, Addendum 1, Tables 1-5, which lists the frequency as 3.1×10^{-3} per year (WSRC 1993a).

- **A6. Criticality During Processing** - The frequency for this reference accident was obtained from WSRC-RP-93-1102, *FB-Line Basis for Interim Operation*, November 1993, Figure 3, which lists a frequency of 1.4×10^{-4} per year (WSRC 1993d).

- **A7. Spill/Liquid Discharge (External)** - The frequency for this reference accident was derived from analyses provided in DOE/EIS-0147, *Continued Operation of K-, L-, and P-Reactors*, December 1990 (DOE 1990), as well as other safety analyses developed for additional SRS facilities. The initiating event is a design basis earthquake with peak horizontal ground accelerations equal to 0.2 times the force of gravity (i.e., $0.2g$) which occurs with an estimated frequency of 2.0×10^{-4} per year, and results in the release of the basin water (3.4 million gallons) to the Savannah River.

- **A8. Spill/Liquid Discharge (Internal)** - The frequency for this reference accident was obtained from DPSTSA-200-10-3, *Receiving Basin for Offsite Fuel (RBOF)*, Addendum 1, Tables 1 - 3, which lists the frequency as 1.1×10^{-1} per year for a representative spill at a hose rack (WSRC 1993a).

A.2.4 Applicability of Accidents to Facilities

This evaluation reviewed Section 1 of the reference document *Technical Data Summary Supporting the Spent Nuclear Fuel Environmental Impact Statement* (WSRC 1994b) to develop a matrix of the selected radiological accidents to the facilities (modules) being considered for the various alternatives and cases. For proposed new facilities, the analysis used best engineering judgment to extrapolate from appropriate accident scenarios based on the descriptions provided in the reference document. Table A-3 lists the connection of facilities to accident scenarios. For example, the Examination and Characterization Facility (module B) identifies a potential accident scenario, A1 (as defined in Table A-2), that should be considered when this facility is utilized to support any case.

Table A-3. Applicable accidents and facilities.

Facility	Module[a]	Accidents
Spent Fuel Receiving, Cask Handling and Fuel Unloading	A	A1
Examination and Characterization	B	A1
Naval Reactor Spent Fuel Examination and Characterization	C	A1, A5, A7, A8
Spent Fuel Repackaging	D	A1, A5, A7, A8
Canister Loading	E	A1, A7, A8
Interim Dry Storage	F	A1, A3
Interim Spent Fuel Storage Pool	G	A1, A5, A7, A8
F-Canyon/F-Area Separations	H, I	A1, A2, A3, A6
H-Canyon/H-Area Separations	J, K, L	A1, A2, A3, A6
Reactor Disassembly Basins	M	A1, A5, A7
Receiving Basin for Offsite Fuels	N	A1, A4, A5, A7, A8

a. As defined in WSRC (1994b).

A.2.5 Facilities and Reference Accidents Associated with each Alternative Case

Table A-4 links alternatives, specific cases, supporting facilities (modules), and accident scenarios. This table identifies the facilities that could be required to support each alternative by specific case. The combined associated accident scenarios for each facility provide the accident spectrum associated with the specific cases for each alternative.

A.2.6 Impacts from Radioactive Release Accidents

This section provides a quantitative discussion of potential consequences to the receptor groups identified in Section A.1.4.1. It also provides a qualitative discussion on potential health effects and consequences for workers at less than 100 meters (328 feet) for each of the potential accident scenarios.

Table A-4. Spent nuclear fuel facilities and accident spectrum by alternatives.

Alternative	Modules[a]	Accidents
1. NO ACTION		
Option 1 - Wet Storage	M, N	A1, A4, A5, A7, A8
2. DECENTRALIZATION		
Option 2a - Dry Storage	B, D, E, F, G, M, N	A1, A3, A4, A5, A7, A8
Option 2b - Wet Storage	B, D, E, G, M, N	A1, A4, A5, A7, A8
Option 2c - Processing	G, H, I, J, K, L, M, N	A1, A2, A3, A4, A5, A6, A7, A8
3. PLANNING BASIS		
Option 3a - Dry Storage	B, D, E, F, G, M, N	A1, A3, A4, A5, A7, A8
Option 3b - Wet Storage	B, D, E, G, M, N	A1, A4, A5, A7, A8
Option 3c - Processing	G, H, I, J, K, L, M, N	A1, A2, A3, A4, A5, A6, A7, A8
4. REGIONALIZATION		
Option 4a - Dry Storage	A, B, D, E, F, G, M, N	A1, A3, A4, A5, A7, A8
Option 4b - Wet Storage	A, B, D, E, G, M, N	A1, A4, A5, A7, A8
Option 4c - Processing	A, G, H, I, J, K, L, M, N	A1, A2, A3, A4, A5, A6, A7, A8
Option 4d - Dry Storage	A, B, C, D, E, F, G, M, N	A1, A3, A4, A5, A7, A8
Option 4e - Wet Storage	A, B, C, D, E, G, M, N	A1, A4, A5, A7, A8
Option 4f - Processing	A, C, G, H, I, J, K, L, M, N	A1, A2, A3, A4, A5, A6, A7, A8
Option 4g - Ship Out	M, N	A1, A4, A5, A7, A8
5. CENTRALIZATION		
Option 5a - Dry Storage	A, B, C, D, E, F, G, H, M, N	A1, A3, A4, A5, A7, A8
Option 5b - Wet Storage	A, B, C, D, E, G, M, N	A1, A4, A5, A7, A8
Option 5c - Processing	A, C, G, H, I, J, K, L, M, N	A1, A2, A3, A4, A5, A6, A7, A8
Option 5d - Ship Out	M, N	A1, A4, A5, A7, A8

a. Source: WSRC (1994b).

A.2.6.1 Radioactive Release Accidents and Consequences for Spent Nuclear Fuel Alternatives. Table A-5 summarizes the information in Tables A-2 through A-4 and provides individual consequences (doses) based on accident type for each case. The table lists consequences for the four receptor groups as follows: Maximum Offsite Individual Dose, the Population to 80 kilometers (50 miles) Dose, the Worker Dose, and the Colocated Worker Dose.

Table A-5. Radioactive release accidents and consequences for spent nuclear fuel alternatives.

Description		Accident	Accident frequency (per year)	Maximally offsite individual dose (rem)	Population to 80 kilometers dose (person-rem)	Worker dose (rem)	Colocated worker dose (rem)
				1. NO ACTION			
Option 1 Wet Storage	A1	Fuel Assembly Breach	1.6×10^{-1}	2.0×10^{-3}	1.7×10^{1}	(a)	1.2×10^{-2}
	A4	Material Release (adjacent facility)	2.4×10^{-3}	6.0×10^{-3}	5.0×10^{1}	(a)	5.0×10^{-2}
	A5	Criticality in Water	3.1×10^{-3}	3.0×10^{-3}	8.8×10^{0}	(a)	1.4×10^{-1}
	A7	Spill/Liquid Discharge (external)	2.0×10^{-4}	5.4×10^{-3}	1.8×10^{1}	(a)	7.6×10^{-2}
	A8	Spill/Liquid Discharge (internal)	1.1×10^{-1}	2.4×10^{-10}	2.0×10^{-6}	(a)	2.0×10^{-11}
				2. DECENTRALIZATION			
Option 2a Dry Storage	A1	Fuel Assembly Breach	1.6×10^{-1}	2.0×10^{-3}	1.7×10^{1}	(a)	1.2×10^{-2}
	A3	Material Release (dry vault)	1.4×10^{-3}	2.1×10^{-6}	6.9×10^{-3}	(a)	(a)
	A4	Material Release (adjacent facility)	2.4×10^{-3}	6.0×10^{-3}	5.0×10^{1}	(a)	5.0×10^{-2}
	A5	Criticality in Water	3.1×10^{-3}	3.0×10^{-3}	8.8×10^{0}	(a)	1.4×10^{-1}
	A7	Spill/Liquid Discharge (external)	2.0×10^{-4}	5.4×10^{-3}	1.8×10^{1}	(a)	7.6×10^{-2}
	A8	Spill/Liquid Discharge (internal)	1.1×10^{-1}	2.4×10^{-10}	2.0×10^{-6}	(a)	2.0×10^{-11}
Option 2b Wet Storage	A1	Fuel Assembly Breach	1.6×10^{-1}	2.0×10^{-3}	1.7×10^{1}	(a)	1.2×10^{-2}
	A4	Material Release (adjacent facility)	2.4×10^{-3}	6.0×10^{-3}	5.0×10^{1}	(a)	5.0×10^{-2}
	A5	Criticality in Water	3.1×10^{-3}	3.0×10^{-3}	8.8×10^{0}	(a)	1.4×10^{-1}
	A7	Spill/Liquid Discharge (external)	2.0×10^{-4}	5.4×10^{-3}	1.8×10^{1}	(a)	7.6×10^{-2}
	A8	Spill/Liquid Discharge (internal)	1.1×10^{-1}	2.4×10^{-10}	2.0×10^{-6}	(a)	2.0×10^{-11}
Option 2c Processing	A1	Fuel Assembly Breach	1.6×10^{-1}	2.0×10^{-3}	1.7×10^{1}	(a)	1.2×10^{-2}
	A2	Material Release (processing)	2.6×10^{-1}	6.8×10^{-5}	5.2×10^{-1}	(a)	9.0×10^{-5}
	A3	Material Release (dry vault)	1.4×10^{-3}	2.1×10^{-6}	6.9×10^{-3}	(a)	(a)
	A4	Material Release (adjacent facility)	2.4×10^{-3}	6.0×10^{-3}	5.0×10^{1}	(a)	5.0×10^{-2}
	A5	Criticality in Water	3.1×10^{-3}	3.0×10^{-3}	8.8×10^{0}	(a)	1.4×10^{-1}
	A6	Criticality in Processing	1.4×10^{-4}	7.0×10^{-3}	8.6×10^{0}	(a)	2.6×10^{-1}

Table A-5. (continued).

Description		Accident	Accident frequency (per year)	Maximally offsite individual dose (rem)	Population to 80 kilometers dose (person-rem)	Worker dose (person-rem)	Colocated worker dose (person-rem)
				2. DECENTRALIZATION			
	A7	Spill/Liquid Discharge (external)	$2.0x10^{-4}$	$5.4x10^{-3}$	$1.8x10^{1}$	(a)	$7.6x10^{-2}$
	A8	Spill/Liquid Discharge (internal)	$1.1x10^{-1}$	$2.4x10^{-10}$	$2.0x10^{-6}$	(a)	$2.0x10^{-11}$
				3. PLANNING BASIS			
Option 3a Dry Storage				Same as Option 2a for Decentralization			
Option 3b Wet Storage				Same as Option 2b for Decentralization			
Option 3c Processing				Same as Option 2c for Decentralization			
				4. REGIONALIZATION			
Option 4a and 4d Dry Storage				Same as Option 2a for Decentralization			
Option 4b and 4e Wet Storage				Same as Option 2b for Decentralization			
Option 4c and 4f Processing				Same as Option 2c for Decentralization			
Option 4g Ship Out				Same as Alternative 1, No Action			
				5. CENTRALIZATION			
Option 5a Dry Storage				Same as Option 2a for Decentralization			
Option 5b Wet Storage				Same as Option 2b for Decentralization			
Option 5c Processing				Same as Option 2c for Decentralization			
Option 5d Ship Out				Same as Alternative 1, No Action			

a. The safety analysis reports from which information was extracted for these accidents were written before the issuance of DOE Orders 5480.23 (DOE 1992); previous orders did not require the inclusion of worker doses.

A.2.6.2 Impacts to Workers at Less than 100 Meters from Radiological Releases.

This section provides a qualitative discussion addressing the impacts due to potential radiological accident scenarios to workers at less than 100 meters (328 feet) involved in SRS spent nuclear fuel management. While worker fatalities may result from release initiators (i.e., plane crashes, seismic event, crane failure, etc.) and not as a direct consequence of a radiation release, this discussion considers only the radiological impacts of an accident, should it occur.

- **A1. Fuel Assembly Breach** - No fatalities to workers would be expected from radiological consequences because the release of the source term would be under water. Attenuation by the water would occur for most products, but the release of noble gases would cause a direct radiation exposure to workers in the area. However, because of the high metallic content of SRS spent nuclear fuel, only a very small fraction of the gases generated in an assembly would be released to the basin water. Air monitors in the area would warn personnel in the event of an airborne release. Timely evacuation would prevent substantial radiation exposures.

- **A2. Material Release (Processing)** - No fatalities to workers would be likely from radiological consequences (Meehan 1995). This scenario assumes that the material released from the process vessels would remain within the Canyon structure and be processed through the Canyon's ventilation and filtration system. Because of shielding effect from the thick concrete walls separating the vessels and areas occupied by workers, the exposures to workers are not expected to be significantly larger than those that would be received during routine operations.

- **A3. Material Release (Dry Vault)** - No fatalities to workers would be likely from radiological consequences. Medium energetic events resulting in the release of radioactive material from the Plutonium Storage Facility vault can result in the dispersal of radioactive materials. For these events, the radioactive material present would bypass the containment and disperse, but would result in a dose well below the lethal level. This assumes that a material release would be distributed into the volume of the smallest room for each unit of operation. It is further assumed that the operator is able to exit the room in 30 seconds (Du Pont 1989). This scenario presumes that the fractions of the plutonium volatized and transported are the same as those applied to the dispersal of the nonvolatile fission products of a criticality. Based on these assumptions, radiological exposure to the worker could occur.

- **A4. Material Release (Adjacent Facility)** - No fatalities to workers would be likely from radiological consequences. The rupture of a waste tank by an explosion could release airborne activity to the shielded cell if the accident occurred during one of the projected 150 times per year when regeneration of the portable columns took place (WSRC 1993a). Although some radiological exposure to the worker could occur, the risk to the worker from the initiating fire and explosion would predominate. Air monitors in the area would warn personnel in the event of an airborne release. Timely evacuation would prevent substantial radiation exposures.

- **A5. Criticality in Water** - No fatalities to workers would be likely from radiological consequences. The use of casks and the underwater handling of spent nuclear fuel greatly reduce the possibility of over-exposure of workers to radiation. The approximately 3 meters (10 feet) of water that covers all fuel provides an attenuation factor of 10^5 for intense gamma radiation and provides protection from direct radiation, even in the event of a criticality. However, a small chance of direct radiation exposure could result due to a floating fuel element or a fuel element inadvertently being raised too high. Strategically located radiation monitors reduce even this probability by alerting workers and sounding an evacuation alarm.

- **A6. Criticality During Processing** - The radiation field generated by a criticality incident could lead to fatalities among workers at the FB-Line facility. As discussed in Section A.2.2, FB-Line inadvertent criticality events are bounding for F- and H-Area spent fuel management processing facilities. This is assumed because workers involved in the FB-Line activities are in close proximity to plutonium metal. Of the 74 personnel that could be present during normal operations, 56 are expected to be within areas which the safety analysis report (WSRC 1993d) identifies as potential criticality accident locations. The shielding due to the concrete floors and walls, the distance between personnel, and the specific nature of the event reduce personnel dose so that only nearby personnel on the floor where the accident occurred would potentially receive a fatal dose. In the event of a criticality accident, DOE estimates that up to 4 deaths could occur, and as many as 50 other workers could receive non-fatal levels of direct radiation.

- **A7. Spill/Liquid Discharge (External)** - No fatalities to workers would be likely from radiological consequences because drainage of the water from the pool or basin would be expected to take several days, or under the most extreme circumstances, several hours, which provides sufficient time for workers to evacuate the area.

- **A8. Spill/Liquid Discharge (Internal)** - No fatalities to workers would be likely from radiological consequences. Minor releases of contaminated water have occurred at the Receiving Basin for Offsite Fuel hose rack platform during the handling of portable deionizers from the reactor areas. One such release was the result of an operator attempting to correct a small leak on a pressurized portable deionizer. The operator was subsequently sprayed with contaminated water, resulting in a radioactive exposure. A spill at the hose rack is not expected to release more than 378.5-liters (100 gallons) of contaminated water.

A.2.7 Point Estimates of Risk

Table A-6 lists the point estimate of risk for each reference accident considered for two receptors. The point estimate of risk is the product of frequency (in occurrences per year) and the number of potential latent fatal cancers. The number of potential latent fatal cancers is the product of dose (in rem for the individual or person-rem for the population) and the ICRP 60 risk factors (4.0×10^{-4} latent fatal cancer per rem for the worker or 5.0×10^{-4} latent fatal cancer per rem for the general public). These point estimates were used to determine the relative risk for each case and to determine the accident that becomes dominant if DOE retires specific facilities during the total period under consideration. For example, all alternatives begin with the immediate storage of spent nuclear fuel in wet pools; however, for the alternative considering interim dry storage, the accident dominating risk will change as the configuration of facilities utilized changes and as spent nuclear fuel or special nuclear material is placed in and remains in interim storage rather than being handled.

A.2.8 Fuel Transition Staging Risk

Table A-7 facilitates the examination of the dominant reference accident during the fuel handling, processing, and storage stages. The use of stages enabled a realistic comparison of risk over the evaluated period. For example, when all fuel has been unloaded, characterized, canned, and put into an interim storage position, consideration of fuel handling events is no longer meaningful.

A.2.9 Adjustment Factors for Comparison Between Alternatives

The accident scenarios described in this document (i.e., Appendix C) differ only slightly between the various alternatives. The scenarios do not account for variations in spent nuclear fuel shipments (including onsite operational transfers) and spent nuclear fuel storage inventories across the alternatives. To provide a realistic comparison across alternatives, DOE developed factors to adjust

Table A-6. Point Estimates of Risk for Reference Accident Scenarios.

Accident Scenario	Descriptions	Frequency (per year)	Potential Fatal Cancers[a]: Maximally Exposed Individual	Potential Fatal Cancers[a]: Population to 80 kilometers	Point Estimate of Risk[b]: Maximally Exposed Individual	Point Estimate of Risk[b]: Population to 80 kilometers
A1	Fuel Assembly Breach	$1.6x10^{-1}$	$1.0x10^{-6}$	$8.5x10^{-3}$	$1.6x10^{-7}$	$1.4x10^{-3}$
A2	Material Release (processing)	$2.6x10^{-1}$	$3.4x10^{-8}$	$2.6x10^{-4}$	$8.8x10^{-9}$	$6.8x10^{-5}$
A3	Material Release (dry vault)	$1.4x10^{-3}$	$1.1x10^{-9}$	$3.5x10^{-6}$	$1.5x10^{-12}$	$4.9x10^{-9}$
A4	Material Release (adjacent facility)	$2.4x10^{-3}$	$3.0x10^{-6}$	$2.5x10^{-2}$	$7.2x10^{-9}$	$6.0x10^{-5}$
A5	Criticality in Water	$3.1x10^{-3}$	$1.5x10^{-6}$	$4.4x10^{-3}$	$4.7x10^{-9}$	$1.4x10^{-5}$
A6	Criticality in Processing	$1.4x10^{-4}$	$3.5x10^{-6}$	$4.3x10^{-3}$	$4.9x10^{-10}$	$6.0x10^{-7}$
A7	Spill/Liquid Discharge (external)	$2.0x10^{-4}$	$2.7x10^{-6}$	$9.0x10^{-3}$	$5.4x10^{-10}$	$1.8x10^{-6}$
A8	Spill/Liquid Discharge (internal)	$1.1x10^{-1}$	$1.2x10^{-13}$	$1.0x10^{-9}$	$1.3x10^{-14}$	$1.1x10^{-10}$

a. ICRP 60 risk factor (5.0×10^{-4}) latent fatal cancer per rem was used to determine potential latent fatal cancers.
b. Units for point estimates of risk are given in potential fatal cancers per year.

Table A-7. Dominant risks based on fuel transition stages.

Fuel/Material Stage	Maximally Exposed Individual Risk	Population to 80 Kilometers Risk
Wet storage	$1.6x10^{-7}$ potential fatal cancer/yr based on accident scenario A1.	$1.4x10^{-3}$ potential fatal cancer/yr based on accident scenario A1.
Dry storage	$1.5x10^{-12}$ potential fatal cancers/yr based on accident scenario A3.	$4.9x10^{-9}$ potential fatal cancers/yr based on accident scenario A3.
Processing (fuel "in-process" by DOE definition)	$1.6x10^{-7}$ potential fatal cancer/yr based on accident scenario A1.	$1.4x10^{-3}$ potential fatal cancer/yr based on accident scenario A1.

frequencies or consequences, depending on the specific circumstances of each alternative. This section describes the methodology and justification used to develop adjustment (scaling) factors for a relative comparison of adjusted point estimates of risk for each alternative on a case-by-case basis.

A.2.9.1 Classification of SRS Accident Scenarios for Applicability to Adjustment Factors. This evaluation screened the SRS accident scenarios to determine which adjustment factor categories were applicable. Table A-8 lists the classification of the different SRS accident scenarios. These adjustment categories are as follows:

- Frequency sensitive due to spent nuclear fuel handling
- Frequency sensitive due to spent nuclear fuel inventories
- Consequence sensitive due to spent nuclear fuel inventories

Table A-8. Adjustment factor classification of SRS accidents.

Accident Scenarios	Accident Description	Frequency Sensitive (Handling)	Frequency Sensitive (Inventory)	Consequence Sensitive (Inventory)
A1	Fuel Assembly Breach	X		
A2	Material Release (Processing)		X	
A3	Material Release (Dry Vault)			X
A4	Material Release (Adjacent Facility)	X		
A5	Criticality in Water	X		
A6	Criticality during Processing		X	
A7	Spill/Liquid Discharge (External)			X
A8	Spill/Liquid Discharge (Internal)			X

The following paragraphs provide the basis for each category selection:

- **A1. Fuel Assembly Breach** - The major initiator for this accident is the mishandling of a fuel assembly. For this reason, the accident frequency for this accident is adjusted to account for the annual number of fuel handling events. The amount of material involved in this accident is limited by the amount of damage that would occur due to the mishandling of a fuel assembly. Therefore, the bounding consequences of this accident are constant and independent of the amount of material available.

- **A2. Material Release (Processing)** - The probability that a release could occur during processing depends on the amount of material that would be processed. Therefore, the accident frequency for this accident is adjusted based on the spent nuclear fuel inventory. Because a maximum amount of material can be processed at any one time, the bounding consequences of this accident are independent of the amount of material on the site.

- **A3. Material Release (Dry Vault)** - The major contributor to the probability of occurrence for this release was external initiators that did not involve material handling. This supports using the same frequency for each alternative. The consequences of this accident are proportional to the amount of material available for release. Therefore, the bounding consequences for this accident are based on the amount of material to be stored.

- **A4. Material Release (Adjacent Facility)** - The initiator for this accident involves the discharge of coolant from a cask into a waste tank. The frequency of occurrence for this accident depends on the number of casks received; therefore, the frequency is adjusted to account for the annual number of fuel shipments.

- **A5. Criticality in Water** - The probability of occurrence of this accident was determined by considering the probability of occurrence of several initiating events. Many of these initiating events involved a criticality due to the mishandling of fuel. Therefore, the frequency for this accident is adjusted to account for the annual number of fuel handling events. The magnitude of the criticality accident is not a function of the amount of material available because the criticality is a highly unlikely, localized event. The consequences for this accident are not adjusted to account for the amount of material available.

- **A6. Criticality During Processing** - The probability that a criticality could occur during processing depends on the amount of material that will be processed. Therefore, the frequency for this accident is adjusted based on the spent nuclear fuel inventory. The magnitude of the criticality accident is not a function of the amount of material available because the criticality is a highly unlikely, localized event. The consequences for this accident are not adjusted to account for the amount of material available.

- **A7. Spill/Liquid Discharge (External)** - The major contributor to the probability of occurrence for this release was external initiators that did not involve material handling. This supports using the same frequency for each alternative. The consequences depend on the amount of fuel in the basin because an increase in the amount of fuel will increase the source term in the basin water. Therefore, the bounding consequences are adjusted for the amount of fuel to be stored.

- **A8. Spill/Liquid Discharge (Internal)** - The major contributor to the probability of occurrence for this release was external initiators that did not involve material handling.

This supports using the same frequency for each alternative. The consequences depend on the amount of fuel in the basin because an increase in the amount of fuel will increase the source term in the basin water. For this reason the bounding consequences are adjusted for the amount of fuel to be stored.

A.2.9.2 Methodology for Determination of Onsite Shipping Frequencies. This section discusses the methodology for determining the onsite shipping frequencies of spent nuclear fuel on a case-by-case basis for each alternative. The annual frequency of handling accidents will vary in direct proportion to the annual number of handling events. However, the consequences of the accident will not vary as a result of spent nuclear fuel handling activities because the amount of material involved in each handling event does not vary. This evaluation assumes that onsite shipments of spent nuclear fuel are near-term shipments, averaged over 5 years. Table A-9 provides a breakdown of current spent nuclear fuel inventories at SRS facilities.

Table A-9. Spent nuclear fuel inventories.[a]

Facility	Number of Aluminum Assemblies[b]	Number of Aluminum Slugs (Buckets[c])	Number of Nonaluminum-Clad Assemblies	Number of Aluminum-Clad Assembly Shipments	Number of Aluminum-Clad Bucket Shipments	Number of Nonaluminum-Clad Assembly Shipments
Receiving Basin for Offsite Fuel (RBOF)	234	107 (2)	261	20	1	22
K-Reactor Basin	1,783	349 (7)	0	149	3	0
L-Reactor Basin	861	13,840 (256)	0	72	86	0
P-Reactor Basin	577	61 (2)	0	48	1	0
Totals	3,455	14,477 (268)	261	289	91	22

a. Basis for inventory numbers: (WSRC 1994c).
b. Assemblies include targets and fuel assemblies. Assembly shipments are based on 12 assemblies per shipment.
c. Number of buckets calculated using 54 slugs per bucket. Bucket shipments are based on 3 buckets per shipment.

A.2.9.2.1 Alternative 1 - No Action – The SRS would send the following number of shipments of aluminum-clad fuel sent to the Receiving Basin for Offsite Fuel from:

- K-Reactor Basin - 152;
- L-Reactor Basin - 158;

- P-Reactor Basin - 49;
- Total - 359 shipments.

All nonaluminum-clad fuel would be sent from the Receiving Basin for Offsite Fuel to a reactor basin (a total of 22 shipments).

The number of shipments would be 380. Because fuel handling would occur at both origin and destination, this number would double (i.e., 760 total shipments). Therefore, over 5 years, this alternative would have an average shipping rate of 152 shipments per year.

A.2.9.2.2 Alternative 2 - Decentralization

- **Option 2a - Dry Storage** - For this option, initial shipments would be the same as those for Alternative 1 (760 shipments at a rate of 152 per year). Subsequent shipments from all storage locations to the new dry storage facilities would total 402 shipments. Because fuel handling would occur at both origin and destination, this number would double (i.e., 804 total shipments). Because all fuel would be moved to dry storage within a 5-year period, this total would have an average rate of 161 shipments per year. Adding all shipments would produce a total of 1,564 shipments at a rate of 313 per year.

- **Option 2b - Wet Storage** - For this option, initial shipments would be the same as those for Alternative 1 (760 shipments at a rate of 152 per year). Subsequent shipments from all storage locations to the new wet storage facilities would total 402 shipments for existing SRS fuel. Because the receipt of offsite fuel would continue prior to the relocation of fuel to the new wet storage facilities, an additional 50 shipments would occur [assuming receipt of five shipments per year of offsite fuel (per Volume 1, Appendix I "Offsite Transportation of Spent Nuclear Fuel")] until 2005. The resulting fuel movement would total 452 shipments. Because fuel handling would occur at both origin and destination, this number would double (i.e., 904 total shipments). Therefore, over 5 years this option would have an average shipping rate of 181 shipments per year. Adding all shipments under this option would produce a total of 1,664 shipments at a rate of 333 per year.

- **Option 2c - Processing** - In this option, all aluminum-clad fuel would move from its present location to the process facilities. All nonaluminum-clad fuel would remain in its present storage locations. The result would be in a total of 380 shipments. As in the previous options, this number would double for a total of 760 shipments. Therefore, over 5 years this option would have an average shipping rate of 152 shipments per year.

A.2.9.2.3 Alternative 3 - Planning Basis

- **Option 3a - Dry Storage** - The movement of materials for this option would be identical to that for Option 2a, resulting in a total of 1,564 shipments at a rate of 313 per year.

- **Option 3b - Wet Storage** - The movement of materials for this option would be identical to that for Option 2b, with the exception of a delay in the receipt of foreign fuel until the new facilities are in operation. This would result in a total of 1,564 shipments at a rate of 313 per year.

- **Option 3c - Processing** - The movement of materials for this option would be identical to that for Option 2c, resulting in a total of 760 shipments at a rate of 152 shipments per year.

A.2.9.2.4 Alternative 4 - Regionalization

- **Option 4a - Dry Storage** - For this option, initial shipments would be the same as Alternative 1 (760 shipments at a rate of 152 per year). Subsequent shipments of the aluminum-clad fuel to the new dry storage facilities would total 380 shipments. (Note: Nonaluminum-clad fuel would be sent offsite from the reactor basins and would not contribute to any further onsite movements.). Because fuel handling would occur at both origin and destination, this number would double (i.e., 760 total shipments). Because all fuel would move to dry storage within about 5 years, this total would have an average shipping rate of 152 shipments per year. Adding all shipments would produce a total of 1,520 shipments at a rate of 304 per year.

- **Option 4b - Wet Storage** - The movement of materials for this option would be identical to that for Option 3b, with the exception of movement of the nonaluminum-clad fuel to the new wet storage facility. This fuel would move off the Site from the reactor basins and would not contribute to any further onsite movements. This would result in a total of 1,520 shipments at a rate of 304 per year.

- **Option 4c - Processing** - The movement of materials for this option would be identical to that for Options 2c and 3c, resulting in a total of 760 shipments at a rate of 152 per year.

- **Option 4d - Dry Storage** - The movement of materials for this option would be identical to those for Options 2a and 3a, resulting in a total of 1,564 shipments at a rate of 313 per year.

- **Option 4e - Wet Storage** - The movement of materials for this option would be identical to that for Option 3b, resulting in a total of 1,564 shipments at a rate of 313 per year.

- **Option 4f - Processing** - The movement of materials for this option would be identical to those for Options 2c, 3c, and 4c, resulting in a total of 760 shipments at a rate of 152 per year.

- **Option 4g - Ship Out** - This option would require the shipping of all spent nuclear fuel at the SRS to a selected regional location. The movement of materials for this option would include the entire spent nuclear fuel inventory at the SRS, resulting in a total of 402 shipments at a rate of 81 per year.

A.2.9.2.5 Alternative 5 - Centralization

- **Option 5a - Dry Storage** - The movement of materials for this option would be identical to those for Options 2a and 3a, resulting in a total of 1,564 shipments at a rate of 313 per year.

- **Option 5b - Wet Storage** - The movement of materials for this option would be identical to that for Option 3b, resulting in a total of 1,564 shipments at a rate of 313 per year.

- **Option 5c - Processing** - The movement of materials for this option would be identical to those for Options 2c, 3c, and 4c, resulting in a total of 760 shipments at a rate of 152 shipments per year.

- **Option 5d - Ship Out** - This option would require the shipping of all spent nuclear fuel at the SRS to a selected central location. The movement of materials for this option would include the entire spent nuclear fuel inventory at the SRS, resulting in a total of 402 shipments at a rate of 81 per year.

A.2.9.3 Methodology for Determination of Offsite Shipping Frequencies. This evaluation determined the total number of offsite shipments using the data contained in Volume 1, Appendix I, "Offsite Transportation of Spent Nuclear Fuel." The total number of Naval Fuel shipments was determined from Table 3 of "Methodology for Adjusting SNF Facility Accident Probabilities and Consequences For Different EIS Alternatives" (dated March 18, 1994).

Naval, foreign, and university shipments would occur throughout the interim management period and could be averaged over the 40-year period covered by this EIS. All other shipments would be averaged over 5 years.

A.2.9.4 Frequency Adjustment Factors for Fuel Handling. For this analysis, DOE assumed the baseline fuel handling rate (events per year) to be the No Action alternative. For the other alternatives, this evaluation divided the expected spent nuclear fuel handling rate by the baseline spent nuclear fuel handling rate (No Action) to obtain the adjustment factor (see Table A-10).

A.2.9.5 Frequency/Consequence Adjustment Factors Due to Inventory. The No Action alternative for the SRS would require the storage of 206 MTHM (227 tons) of fuel. Using this amount as the baseline, this evaluation compared the amount of fuel for the other alternatives to the base number, as listed in Table A-11. These adjustment factors can be applied to either a frequency or a consequence, depending on the classification of the accident scenario as listed in Table A-8.

A.3 Chemical Hazard Evaluation

A.3.1 Selection of Reference Chemical Hazard

A review of the same safety analyses used to generate the spectrum of radiological accident scenarios failed to identify a quantitative discussion of chemical hazards. However, each of the safety analyses provided a qualitative discussion of chemical hazards. Thus, Section 5.15.3 discusses chemical hazards associated with existing spent nuclear fuel facilities qualitatively. This qualitative evaluation was determined to be appropriate based on three criteria: sliding scale in proportion to significance, public perception of severity, and long-term effects of chemicals not known. For completeness, a separate risk assessment (WSRC 1993c) provided a quantitative discussion of chemical hazards for the Receiving Basin for Offsite Fuel facility. This assessment described a bounding chemical hazard accident involving the release of nitrogen dioxide vapor.

Table A-10. Fuel handling frequency adjustment factors.

Option Number	Estimated Annual Shipping Rate	Frequency Adjustment Factor
Alternative 1 - No Action		
Option 1	152	Baseline
Alternative 2 - Decentralization		
Option 2a	316	2.08
Option 2b	333	2.19
Option 2c	157	1.03
Alternative 3 - Planning Basis		
Option 3a	375	2.47
Option 3b	375	2.47
Option 3c	216	1.42
Alternative 4 - Regionalization		
Option 4a	421	2.77
Option 4b	421	2.77
Option 4c	269	1.77
Option 4d	394	2.59
Option 4e	394	2.59
Option 4f	234	1.54
Option 4g	160	1.05
Alternative 5 - Centralization		
Option 5a	803	5.28
Option 5b	803	5.28
Option 5c	643	4.23
Option 5d	160	1.05

Table A-11. Inventory adjustment factors for each alternative.

Alternative	Inventory[a] (MTHM[b])	Adjustment Factor
No Action	206.27	Baseline
Decentralization	219.89	1.07
Planning Basis	222.76	1.08
Regionalization - A	213.09	1.03
Regionalization - B	256.62	1.24
Centralization	2,741.80	13.30

a. Source: Wichmann (1995).
b. Metric Tons Heavy Metal; to convert to tons, multiply by 1.1023.

A.3.2 Hazardous Chemical Inventories

The inventory of hazardous chemicals at each facility was determined by using the "Savannah River Site Tier Two Emergency and Hazardous Chemical Inventory Report" (WSRC 1994a) to get the facility's total chemical inventory, then listing those chemicals that also appeared on the EPA's "List of Lists" (EPA 1990). The chemical inventories listed in Tables A-12 through A-15 represent facilities used for wet storage and/or processing of spent nuclear fuel. The SRS maintains no large-scale dry storage facilities; thus, chemical inventories for dry storage facilities are not listed.

Table A-12. Hazardous chemical inventory for the Receiving Basin for Offsite Fuel.

Chemical	Maximum Daily Amount (Kg)[a]	Average Daily Amount (Kg)
Ethylene glycol	2,981	23
Methyl ethyl ketone	2	2
Nitric acid	4,731	2,365
Phosphoric acid	3,953	3,953
Sodium hydroxide (caustic soda)	5,800	2,900
Sodium nitrite	3,070	1,535

a. To convert kilograms to pounds, multiply by 2.2046.

Table A-13. Hazardous chemical inventory for the reactor basins (typical).

Chemical	Maximum Daily Amount (Kg)[a]	Average Daily Amount (Kg)
Aluminum sulfate (solution)	570	230
Ethylene glycol (thermal arc torch coolant concentrate)	2	2
Hydrogen peroxide	1	1
Nitric acid	75	75
Sodium hydroxide	454	454
Sodium hypochlorite	11	6
Zinc	0.5	0.5

a. To convert kilograms to pounds, multiply by 2.2046.

Table A-14. Hazardous chemical inventory for H-Area.

Chemical	Maximum Daily Amount (Kg)[a]	Average Daily Amount (Kg)
Dichlorodifluoromethane (Freon 12)	227	68
Dichlorodifluoromethane (Racon 12)	227	0
Ethylene glycol	4.0	2.0
Hydrofluoric acid	1	0.5
Hydrogen peroxide	0.5	0.0
Mercury	4,900	4,900
Methyl ethyl ketone	3	3
Nitric acid	10	5
Nitric oxide	1,300	1,300
Phosphorus pentoxide	1	1
Potassium permanganate (Cairox)	200	100
Sodium hydroxide	1	1
Sodium hypochlorite	41	29
Sulfuric acid	1	0.5
Trichlorofluoromethane (Freon 11)	1,150	1,000
Trichlorofluoromethane (Genetron 11)	450	0

a. To convert kilograms to pounds, multiply by 2.2048.

Table A-15. Hazardous chemical inventory for F-Area.

Chemical	Maximum Daily Amount (Kg)[a]	Average Daily Amount (Kg)
Dichlorodifluoromethane (Freon 12)	1	0.5
Dichlorodifluoromethane (Racon 12)	1	0
Ethylene glycol	4	2
Hydrofluoric acid	1,177	1,177
Potassium permanganate	3	1
Sodium hydroxide	0.5	—
Sodium hypochlorite	7	4
Sulfuric acid	30	—
Trichlorofluoromethane (Freon 11)	900	450

a. To convert kilograms to pounds, multiply by 2.2048.

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