DOE/EIS-0218F
February 1996
Volume 1

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Department of Energy
Washington, DC 20585

February 8, 1996

Dear Interested Party:

I am enclosing a copy of the final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel. The Department of Energy, in cooperation with the State Department, prepared the final Environmental Impact Statement.

This study analyzes the potential environmental impacts of adopting a policy to manage foreign research reactor spent fuel containing uranium enriched in the United States. In particular, the study examines the comparative impacts of several alternative approaches to managing the spent fuel. The analyses demonstrate that the impacts on the environment, workers and the general public of implementing any of the alternative management approaches would be small and within applicable Federal and state regulatory limits.

The Department's preferred approach to managing the spent fuel, referred to in the study as the "preferred alternative," is for the Department to receive the spent fuel into the United States, and to manage it at the Department's Savannah River Site in South Carolina and the Idaho National Engineering Laboratory. The spent fuel would be shipped to the United States over 13 years through two military ports. The Charleston Naval Weapons Station in South Carolina would receive about one to two shipments every month beginning in 1996. The Concord Naval Weapons Station in California would receive far fewer shipments (as few as five shipments over a 13-year period) beginning in 1997.

The final Environmental Impact Statement is a three-volume document, approximately 4000 pages in length. Volume 1 (494 pages) describes the policy considerations of adopting a policy to manage foreign research reactor spent fuel, and the potential environmental impacts. Volume 2 (1111 pages) contains eight appendices relating to the technical analyses. Volume 3 (2230 pages) contains the public's comments on the draft Environmental Impact Statement, the Department's responses to those comments, and summaries of the 17 public hearings held throughout the United States during the 90-day comment period on the draft.

If you would like another copy of the entire study, a particular volume, or an additional copy of the Summary, we would be pleased to send it to you. Please let us know by calling the Department's Center for Environmental Management Information at 1-800-736-3282 (toll-free). The entire document will be placed in the public reading rooms and information locations listed in the Summary.

The Department will not make a final decision on whether to adopt the proposed policy until late March 1996. Thank you for your interest in this proposed action.

Sincerely,

Thomas P. Grumbly

Thomas P. Grumbly
Assistant Secretary for
Environmental Management

Enclosure

DOE/EIS-0218F
February 1996

Volume 1

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Cover Sheet

Responsible Agencies: Lead Agency: United States Department of Energy
Cooperating Agency: United States Department of State

Title: Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Contact: For further information, concerning this Final Environmental Impact Statement, contact:

Charles Head, Program Manager
Office of Spent Nuclear Fuel Management (EM-67)
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585

For general information on the United States Department of Energy's National Environmental Policy Act process, call 1-800-472-2756 to leave a message, or contact:

Carol Borgstrom, Director
Office of NEPA Policy and Assistance (EH-42)
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585
202-586-4600

Abstract: The United States Department of Energy and United States Department of State are jointly proposing to adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel containing uranium enriched in the United States would be covered by the proposed policy. The purpose of the proposed policy is to promote U.S. nuclear weapons nonproliferation policy objectives, by seeking to reduce and eventually eliminate highly-enriched (weapons-grade) uranium from civilian commerce worldwide. Environmental effects and policy considerations of three Management Alternative approaches for implementation of the proposed policy are assessed. The three Management Alternatives analyzed are: (1) acceptance and management of the spent nuclear fuel by the Department of Energy in the United States, (2) facilitate the management of the spent nuclear fuel at one or more foreign facilities (under conditions that satisfy United States nuclear weapons nonproliferation policy objectives), and (3) a combination of elements from one or both of Management Alternatives 1 and 2 (Hybrid Alternative). A No Action Alternative is also analyzed.

For each Management Alternative, there are a number of implementation alternatives. For Management Alternative 1, this document addresses the environmental effects of various implementation alternatives, such as varied policy durations, management of various quantities of spent nuclear fuel, chemical separation, developmental treatment and/or packaging technologies, and differing financing arrangements. Environmental impacts are also examined at various potential ports of entry, along truck and rail transportation routes, at candidate management sites, and for alternate storage technologies. For Management Alternative 2, this document addresses the environmental effects of two implementation alternatives: (1) assisting foreign nations with storage; and (2) assisting foreign nations with reprocessing

of the spent nuclear fuel. With respect to Management Alternative 3, an example Hybrid Alternative is analyzed wherein a portion of the spent nuclear fuel would be processed at overseas facilities and the remaining portion would be managed in the United States.

The United States Department of Energy and United States Department of State, in consultation with other government agencies, designate the acceptance and management of the foreign research reactor spent nuclear fuel in the United States (i.e., Management Alternative 1 with modifications to several basic implementation elements) as the preferred alternative.

Public Comments: The public comment period on the Draft EIS was conducted from April 21, 1995 to July 20, 1995. During this period, DOE held 17 public hearings in the locations most likely to be directly affected by the EIS alternatives, including the 10 candidate ports of entry and 5 candidate spent nuclear fuel management sites. In addition, a public hearing was held in Washington, D.C. The Draft EIS was made available to the public through mailings, requests to DOE's Environmental Management Information Center, and at DOE Public Reading Rooms and other designated information locations.

Foreword

This Final Environmental Impact Statement presents an evaluation of policy considerations and potential environmental impacts resulting from the U.S. Department of Energy and the U.S. Department of State joint proposal to adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel that contains uranium enriched in the United States would be covered by the proposed policy. The purpose of the proposed policy would be to promote nuclear weapons nonproliferation objectives of the United States, specifically by seeking to reduce, and eventually to eliminate, highly-enriched (weapons-grade) uranium from civil commerce worldwide. This policy is jointly proposed by the U.S. Department of Energy and the U.S. Department of State. This document was prepared in compliance with the National Environmental Policy Act and in accordance with regulations issued and published by the Council on Environmental Quality (40 CFR Parts 1500-1508) and the U.S. Department of Energy (10 CFR Part 1021).

Environmental effects and policy considerations of several alternative approaches for implementation of the proposed policy are assessed. Three Management Alternatives are analyzed: (1) acceptance and management of the spent nuclear fuel by the Department of Energy in the United States; (2) facilitate the management of the spent nuclear fuel at one or more foreign facilities under conditions that satisfy United States nuclear weapons nonproliferation policy objectives; and (3) a combination of components of Management Alternatives 1 and 2 (Hybrid Alternative Example). A No Action Alternative is also analyzed.

For each Management Alternative, there are a number of alternatives for its implementation. For Management Alternative 1, this document addresses the policy implications and environmental effects of various implementation alternatives such as varied policy durations, management of various quantities of spent nuclear fuel, and differing financing arrangements. Environmental impacts at various potential ports of entry, along truck and rail transportation routes, at candidate management sites, and for alternate storage technologies are also examined. For Management Alternative 2, this document addresses two subalternatives: (1) assisting foreign nations with storage; and (2) assisting foreign nations with reprocessing of the spent nuclear fuel. With respect to Management Alternative 3, a hybrid alternative example is analyzed, utilizing the analysis provided for Management Alternatives 1 and 2, wherein a portion of the spent nuclear fuel would be processed at overseas facilities and the remaining portion would be managed in the United States.

A Notice of Intent to prepare this document was published in the Federal Register on October 21, 1993. Nine public scoping meetings were conducted during November and December of 1993. The period for acceptance of public comments on this document closed on December 8, 1993. However, the United States Department of Energy continued to accept written comments through January 31, 1994. In October 1994, the Implementation Plan for this Environmental Impact Statement was issued to provide guidance for its preparation and to record the U.S. Department of Energy's disposition of comments received during the scoping process.

The Draft Environmental Impact Statement was issued in April 1995. The public comment period on the Draft Environmental Impact Statement was from April 21, 1995 to July 20, 1995. During this period, DOE held 17 public hearings in the locations most likely to be directly affected by the EIS alternatives, including the 10 candidate ports of entry and 5 candidate spent nuclear fuel management sites. In addition, a public hearing was also held in Washington, D.C. The Draft EIS was made available to the public through mailings, requests to DOE's Environmental Management Information Center, and at DOE Public Reading Rooms and other designated information locations.

Results of the environmental analyses are presented in two volumes. Volume 1 is composed of eight chapters. Chapter 1 gives the background description of the United States nuclear weapons nonproliferation policy and describes the purpose and need for the proposed action. Chapter 2 then states the proposed policy and describes the three Management Alternatives for its implementation. It includes a discussion of the basic implementation components of Management Alternative 1, as well as implementation alternatives that vary one component of the basic implementation of Management Alternative 1. The implementation alternatives include variations on the duration of the policy, alternative amounts of material that might be covered by the policy, and various financing alternatives. The potential ports of entry, transportation routes, candidate spent nuclear fuel management sites and storage technologies are also described. This chapter also describes Management Alternative 2, which contains two subalternatives for its implementation. Subalternative 1 is to provide assistance to foreign nations with storage of the spent nuclear fuel. Subalternative 2 is to provide assistance with reprocessing of the spent nuclear fuel at one or more foreign locations. Management Alternative 3 is also discussed in this Chapter by tiering off the evaluation and analyses provided for Management Alternatives 1 and 2. The potentially affected environment under Management Alternatives 1 and 3 is described in Chapter 3. Essential results of the environmental analyses are then given in Chapter 4, which summarizes the methods used in the evaluation and provides an assessment of the environmental effects. Details of the environmental analyses are provided in the appendices, which comprise Volume 2 of this document. Chapter 5 describes applicable laws, regulations, and other requirements. A list of the preparers of this Final Environmental Impact Statement, agencies consulted, and references are provided in Chapters 6, 7, and 8, respectively. In addition to these two volumes, a Volume 3 (Comment Response Document) has been added to the Final Environmental Impact Statement which contains the written and oral comments received during the public comment period for the Draft Environmental Impact Statement.

In consideration of public comments, DOE has added information to the EIS including: clarification of the proposed U.S. policy on accepting spent nuclear fuel from allies; examination of the consequences of sabotage or terrorist attack; safety of transportation casks; re-examination of the shipboard fire analysis, and general provisions of transportation and emergency response regulations and management. The Naval Weapons Station at Charleston was analyzed in addition to the other terminals of the Port of Charleston within the greater Charleston area that were discussed in the Draft Environmental Impact Statement.

This Final Environmental Impact Statement has a two-fold purpose. The first purpose is to provide decision makers in the U.S. Department of Energy and the U.S. Department of State with an evaluation of the environmental effects of these policies. The second purpose is to inform the public concerning the essential features, policy considerations, and potential environmental effects of the proposed policy, and to provide the public an opportunity to provide feedback to the U.S. Department of Energy and the U.S. Department of State on the proposed policy.

Reader's Guide

In response to comments submitted after issuance of the Draft Environmental Impact Statement in April 1995, and due to additional technical and policy details not available at the time of issuance of the Draft Environmental Impact Statement, Volumes 1 and 2 of the Final Environmental Impact Statement contain revisions and changes. The revisions and changes made since issuance of the Draft Environmental Impact Statement are indicated by a line in the margin of Volumes 1 and 2. A new Appendix H has been added to Volume 2 to describe the general provisions associated with transportation planning for potential shipments of foreign research reactor spent nuclear fuel. In addition, Volume 1 and each appendix in Volume 2 provide a unique reference list to enable the reader to further review and research selected topics. The U.S. Department of Energy has established reading rooms and information locations across the United States where these references may be reviewed or obtained for review through interlibrary loan. The addresses and phone numbers for these reading rooms and information locations are provided at the end of the accompanying Summary.

Table of Contents

Volume 3

List of Figures

List of Tables

Acronyms and Abbreviations

BNFP	Barnwell Nuclear Fuels Plant
CFR	Code of Federal Regulations
Ci	Curie
cm	centimeter
DOE	Department of Energy
EDE	Effective Dose Equivalent
EIS	Environmental Impact Statement
E-MAD	Engine Maintenance and Disassembly
FAST	Fluorinel Dissolution and Fuel Storage
FMEF	Fuel Maintenance and Examination Facility
g	gram
ha	hectare
HEU	Highly-Enriched Uranium
ICPP	Idaho Chemical Processing Plant
IFSF	Irradiated Fuel Storage Facility
ISO	International Organization for Standardization
kgTM	kilograms of Total Mass
km	kilometer
l	liter
LCF	latent cancer fatality
LEU	Low Enriched Uranium
m	meters
MACCS	MELCOR Accident Consequences Code System
MEI	Maximally Exposed Individual
mg	milligram
mg/l	milligrams per liter
mi	mile
min	minute
ml	milliliter
mm	millimeter
MOTSU	Military Ocean Terminal at Sunny Point
mrem	millirem
MTHM	Metric Tons of Heavy Metal
MTR	Material Test Reactor
NEPA	National Environmental Policy Act
NPAI	Nearest Public Access Individual
NRC	Nuclear Regulatory Commission
NWS	Naval Weapons Station
ppt	parts per thousand
rad	radiation absorbed dose
RBOF	Receiving Basin for Offsite Fuels
rem	roentgen equivalent man
RERTR	Reduced Enrichment for Research and Test Reactors
SNF&INEL Final EIS	Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement
TRIGA	Training, Research, Isotope, General Atomic reactors

1. Introduction

Reducing the threat of the proliferation of nuclear weapons is one of the foremost goals of the United States. Proper management of spent nuclear fuel from foreign research reactors is essential to the efforts aimed at achieving these goals since much of this fuel contains highly-enriched uranium[1] (HEU), which can be directly used in simple nuclear weapons.

The concern over appropriate management of foreign research reactor spent nuclear fuel was reiterated in the Presidential Directive on Nonproliferation and Export Controls, issued by President Clinton on September 27, 1993. In particular, the Presidential Directive included the following language: "We will also seek to minimize the use of highly-enriched uranium in civil nuclear programs. To this end, the Secretary of Energy will review the need for programs to develop alternative fuels for research reactors and accelerate steps towards implementation of a policy of taking back U.S.-origin spent fuels from foreign research reactors."

1.1 Policy Background

Since 1945, every U.S. Administration has recognized that preventing the further spread of nuclear weapons must be a fundamental national security and foreign policy objective of the United States. The initial U.S. approach to nuclear technology was to classify all nuclear activities. However, the United States soon realized that it would be impossible to prevent other nations from acquiring nuclear technology.

Consequently, since the 1950's, beginning with the "Atoms for Peace" program, the United States has provided peaceful nuclear technology to foreign nations in exchange for their promise to forego development of nuclear weapons. In addition, the United States requires that any nuclear technology provided shall be subject to international safeguards and inspections to prevent diversion of materials or technology to nuclear weapons activities.

The Atomic Energy Act of 1946 was the first U.S. legislation regulating nuclear activities. The Act prohibited international nuclear cooperation until effective international safeguards were in place to prevent such cooperation from assisting in the development of foreign nuclear weapons programs. A major revision of the Atomic Energy Act in 1954 provided that foreign countries receiving nuclear assistance had to accept conditions on its use, including making a pledge not to use nuclear materials or equipment provided by the United States for military purposes.

Simply put, peaceful nuclear cooperation, under international safeguards, has been a critical component of U.S. nuclear weapons nonproliferation policy since the beginning of the atomic age. The intent of such peaceful nuclear cooperation is to prevent the development of nuclear weapons programs worldwide.

A major element of the "Atoms for Peace" program for peaceful nuclear cooperation, particularly in the early years, was the provision of research reactor technology and the HEU necessary to fuel the research reactors. Research reactors play a vital role in important medical, agricultural, and industrial applications,

1 *Uranium enriched to 20 percent or greater in isotope 235 is known as HEU.*

and also provide a tool for fundamental scientific research. For example, research reactors are a vital tool in cancer therapy and radioimmunoassay blood testing. There are approximately 30,000 medical procedures per day in North America, 8,000 to 10,000 procedures per day in Europe, and 8,000 to 10,000 procedures per day on other continents using medical isotopes produced in research reactors in other countries. Neutron radiography provided by research reactors has enabled researchers to diagnose defects in metals and engines of many varieties and to conduct research on new materials, computer chips, and chemicals. Radioisotopes produced in research reactors have been used in leak detection in industrial components and equipment, aluminum production, and semiconductor and solar panel research. Neutron scattering experiments done in research reactors have provided insight into the biostructure of organic substances and have advanced the development of magnetic and superconducting materials. Research reactors have also been used in the environmental sciences to study waste migration, mine drainage, diffusion and transport of pollutants, water chemistry, sediment transport, atmospheric dispersion, and toxic waste management. Another important use of research reactors is irradiation testing of materials and fuel forms, including safety experimentation, to support advanced fuel design and waste management development for use in the power industry. Research reactors also have served as major training facilities in nuclear technology. For example, the research reactor operating in Austria is used by the International Atomic Energy Agency to train personnel who conduct international inspections of weapons and civil nuclear facilities worldwide.

The transfer of enriched uranium from the United States to other nations under the "Atoms for Peace" program was usually supported by a bilateral research agreement for each foreign research reactor. Before 1964, these agreements provided for the lease of the enriched uranium, with explicit provision for the return of the spent nuclear fuel to the United States. After 1964, most agreements provided for the sale of this material to the foreign nation.

After its use (irradiation) in a research reactor, the used (spent) fuel was generally returned to the United States where it was reprocessed to extract the uranium still remaining in the spent fuel. In this way, the United States maintained control over the HEU, which otherwise could be used in the production of nuclear weapons. The United States began accepting HEU spent nuclear fuel from foreign research reactors in 1958.

After 1964, the operative policy under which the United States accepted foreign research reactor spent nuclear fuel containing uranium enriched in the United States became known as the "Off-Site Fuels Policy." This policy was implemented through a series of *Federal Register* Notices issued until 1987, and was incorporated into bilateral international agreements with recipient countries. The term "Off-Site Fuels Policy" was used to indicate that the spent nuclear fuel had been irradiated at facilities not owned by the Department of Energy (DOE). Under the "Off-Site Fuels Policy," the United States accepted, temporarily stored, and reprocessed spent nuclear fuel containing HEU enriched in the United States. The rationale for the policy was to discourage the stockpiling abroad of spent nuclear fuel containing HEU and to recover the fuel value of the HEU remaining in the spent nuclear fuel.

In response to increasing congressional and public concern about the potential diversion of HEU for use in nuclear weapons by foreign nations, subnational groups, or terrorist organizations, DOE in 1978 initiated the Reduced Enrichment for Research and Test Reactors (RERTR) program. The RERTR program was aimed at reducing the use of HEU in civilian programs by promoting the conversion of foreign research reactors from HEU fuel to low enriched uranium (LEU) fuel. Research reactors are of particular interest in this endeavor because the major civilian use of HEU is as fuel in nuclear research reactors.

As a part of the RERTR program, DOE developed LEU fuel and worked with foreign research reactor operators to modify their reactors to run on such fuel. The foreign research reactor operators who converted to LEU fuel did so in support of nuclear weapons nonproliferation objectives, even though such conversions were expensive and generally resulted in reductions in the capabilities of the reactors and increased operating costs.

From the beginning of the RERTR program, foreign research reactor operators made it clear that their willingness to convert their research reactors to LEU fuel was contingent upon the continued acceptance by DOE of their spent nuclear fuel for disposition in the United States. In 1986, to further encourage foreign research reactor operators to convert to LEU fuel, the DOE "Off-Site Fuels Policy" was extended to include the acceptance of spent nuclear fuel containing LEU enriched in the United States.

The RERTR program has been highly successful and many foreign research reactors have been modified to operate, or have been designed to operate, with the high-density LEU fuels developed by the RERTR program, instead of HEU fuel. Of the 42 foreign research reactors with power levels equal to or above one million watts that use U.S. enriched fuel, 37 could operate with the currently available high-density LEU fuels. Of these, 25 are either operating on LEU fuel, or have ordered LEU fuel, and DOE anticipates that an additional eight reactors will convert to LEU fuel by 2001. Work is underway to develop improved high-density LEU fuels that would enable the remaining HEU fueled reactors to convert as well. Thus, the RERTR program has contributed to a significant reduction in the use of HEU in foreign research reactors.

The RERTR program is also developing the technology necessary to substitute LEU for the HEU in targets that are currently irradiated in reactors to produce the radioisotope molybdenum-99 for use as a diagnostic tool in nuclear medicine. The current limited nuclear commerce in HEU for medical targets can be reduced and eventually eliminated when LEU targets become available. When combined, these RERTR program activities can virtually eliminate the need for civilian commerce in HEU.

In 1988, DOE's "Off-Site Fuels Policy" to accept HEU spent nuclear fuel expired. At the end of 1992, the policy as it applied to the acceptance of LEU spent nuclear fuel also expired. The "Off-Site Fuels Policy" was not immediately renewed because of the need to assess the environmental impacts of a new policy. Because the United States has not been in a position to accept HEU fuel for 6 years [except for two recent "urgent relief" shipments of 252 spent nuclear fuel elements, 153 of which were from Denmark, Austria, Sweden, and the The Netherlands, with the remaining 99 elements from Switzerland and Greece, conducted under DOE's "Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel" (DOE, 1994m)], many foreign research reactor operators will soon run out of storage capacity or face safety and regulatory issues associated with the presence of spent nuclear fuel at their sites.

Although those foreign research reactors who could obtain regulatory clearance to build new storage capacity could do so within the duration of the proposed policy, they do not have time to do so in the near term before they run out of space. Storage of the spent nuclear fuel is a concern because it contains both enriched uranium (some of it highly-enriched material suitable for use in nuclear weapons) and highly radioactive waste products. The storage of such spent nuclear fuel must be accomplished with considerable care to ensure that the spent nuclear fuel does not corrode. If it does corrode, it could release fission products within the storage facility (making action to protect the spent nuclear fuel from further degradation difficult). In the extreme, uranium could be released from the spent nuclear fuel and settle to the bottom of the storage facility, creating the potential for a chain reaction.

Even if the spent nuclear fuel is kept in pristine condition (a relatively straightforward task given the resources and determination to store it correctly), the accumulation of large quantities of spent nuclear fuel containing HEU raises the possibility that some of the spent nuclear fuel might be stolen and its uranium diverted into a nuclear weapons program. In addition, spent nuclear fuel storage is an expensive undertaking (partially so because of the steps needed to avoid the problems outlined above) and is limited by local regulation in many countries. As a result, the cessation of the U.S. acceptance of foreign research reactor spent nuclear fuel associated with the expiration of the "Off-Site Fuels Policy" has created significant problems for the research reactor operators and has undercut the perceived reliability of the United States as a partner in peaceful nuclear cooperation, a cornerstone of U.S. nuclear weapons nonproliferation commitments enshrined in the *Treaty on the Non-Proliferation of Nuclear Weapons.*

With respect to the broader role that the United States plays in worldwide peaceful nuclear cooperation, President Reagan warned as early as July 1981, that "if we are not such a partner, other countries will tend to go their own ways and our influence will diminish. This would reduce our effectiveness in gaining the support we need to deal with proliferation problems." (Statement on the U.S. Nuclear Nonproliferation Policy, July 16, 1981.) More recent correspondence from the United States National Security Council, Department of State, Department of Defense, and the Nuclear Regulatory Commission (NRC) underscores the importance and urgency of support for the RERTR program objectives and the need for immediate action to reduce the use of HEU in civil programs. For example, in a recent letter to Secretary of Energy Hazel R. O'Leary, Secretary of State Warren Christopher stated, "The spent fuel acceptance policy which the EIS supports is central to our goal of preventing the spread of nuclear weapons -- and therefore to a major national security objective of this administration" (see Appendix G).

If the United States does not accept the foreign research reactor spent nuclear fuel, some of the foreign research reactors may be forced to shut down, as they will have no way to store any additional spent nuclear fuel. Other research reactor operators may have the option of reprocessing[2] their spent nuclear fuel (separating the uranium from the fission products for use as new fuel) at existing facilities. British and/or French reprocessing plants might accept foreign research reactor spent nuclear fuel for reprocessing, but have done so in the past only on the condition that the reprocessing customer agrees to take back the reprocessing wastes. Some of the countries in which the foreign research reactors are located do not have a domestic waste repository or other facilities for storing the highly-radioactive wastes generated from reprocessing for the near term. Other countries will not allow reprocessing because they object to reprocessing on environmental or nuclear weapons nonproliferation grounds.

While the U.S. Government has full confidence in the physical protection and safeguards systems in place at the British and French reprocessing facilities, reprocessing of spent nuclear fuel containing HEU, as it has been done in the past, would sustain international commerce in HEU, in direct contradiction to the U.S. position on nuclear weapons nonproliferation. It would likely mean that the research reactors pursuing this option would continue operations on the HEU fuel cycle because currently they have a method of disposing of HEU spent nuclear fuel, but not LEU spent nuclear fuel. Neither Dounreay (the British reprocessing facility in Scotland), Marcoule in France, nor any other available facility is currently accepting or has the special equipment that would be used in the United States to reprocess the high density LEU fuels that the United States is encouraging foreign research reactors to use to replace the

2 *Reprocessing refers to the disassembly of spent nuclear fuel (usually by dissolving it in acid) to allow the uranium, and possibly other fissile materials, to be separated from the fission products and structural parts of the spent nuclear fuel. The fissile materials can then be reused, and the fission products are discharged as waste. These wastes are dissolved in specially formulated molten glass and cast into stainless steel cylinders (or in some cases, in foreign countries, may be mixed with special concretes and poured into steel drums) prior to disposal.*

HEU fuels. Hence, if the research reactors decide to use reprocessing as it has been managed up to this point to prevent backlogs of spent nuclear fuel from building up, they would have to continue to use HEU fuels. This could result in reactor operators delaying or canceling plans to convert to LEU, or in some cases, withdrawing from the RERTR program and reconverting from LEU to HEU fuels. The United Kingdom Atomic Energy Authority is considering adding an LEU processing capability to its plant in Dounreay. At this time, it is not clear that this plant will continue to operate. If the Dounreay facility continues to operate and if an LEU reprocessing capability is installed, that would mean that Dounreay customers could operate their research reactors on LEU.

If some foreign research reactor operators were to withdraw from the RERTR program and rely instead on HEU fuels, with attendant lower costs and enhanced performance, other research reactor operators would be under pressure to convert to the use of HEU for competitive reasons. Since the United States, under the Energy Policy Act of 1992, is barred from exporting HEU to virtually all foreign research reactors, research reactor operators seeking continued use of HEU would be forced to seek alternate suppliers. Russia and China are sources of HEU; and, should they choose to provide a ready supply of HEU, many foreign research reactor operators would be forced to consider abandoning the RERTR program and reconverting to HEU. In addition, the United States is currently attempting to convince Russia and China to implement programs similar to the United States' RERTR program to encourage their nuclear fuel customers to phase out use of HEU in their research reactors. However, if the United States cannot convince those countries to which it has exported nuclear fuel to stop using HEU, the United States would stand little chance of convincing Russia and China to do so with the countries to which they export nuclear fuel.

Additionally, several developed countries involved in the RERTR program are exporters of research reactors. In recent years, they have required that reactors exported to other countries be fueled with LEU. However, if foreign research reactor operators begin delaying or canceling plans to convert to LEU, and thereby continue to use HEU, foreign research reactor purchasers would demand HEU-fueled reactors. This could lead to renewed international commerce in weapons-usable HEU, and would be antithetical to the policy goal of seeking to minimize and eventually eliminate the civil use of HEU.

Another crucial consideration in proposing to accept spent nuclear fuel shipments from foreign research reactors is the *Treaty on the Non-Proliferation of Nuclear Weapons*. The Non-Proliferation Treaty is the basis for the world's nuclear weapons nonproliferation regime. The purpose of the Non-Proliferation Treaty is to keep the number of countries with nuclear weapons from growing. Five countries acknowledge having nuclear weapons: the United States, Russia, the United Kingdom, France, and China. In addition to the five acknowledged nuclear weapons States, 175 other nonnuclear weapons States are members of the treaty. The obligations for compliance with the Non-Proliferation Treaty apply to both nuclear weapons States and nonnuclear weapons States. While nonnuclear weapons States agree not to pursue development or acquisition of nuclear weapons or other nuclear explosive devices, the nuclear weapons States commit to the eventual elimination of their nuclear weapons arsenals and to assist nonnuclear weapons States with peaceful applications of nuclear energy. The Off-Site Fuels Policy and RERTR program are examples of how the United States has helped other nations with peaceful applications of nuclear energy in the past.

The parties to the Non-Proliferation Treaty met in May of 1995 and agreed by consensus to extend the treaty indefinitely and without conditions. Making this vital treaty a permanent part of the international nonproliferation regime was an important U.S. foreign policy achievement. One key to the success of the 1995 Non-Proliferation Treaty Conference was the ability of the United States to convince other

Non-Proliferation Treaty parties that the nuclear weapons States were in compliance with their obligations under Article IV of the Non-Proliferation Treaty to assist the nonnuclear weapons States with applications of nuclear energy for peaceful purposes.

Although the Non-Proliferation Treaty was extended indefinitely at the 1995 Non-Proliferation Treaty Conference, the parties also agreed to review the treaty every five years to ensure that all parties are in compliance. Any country which has been compelled to shut down its research reactors, or has been forced to seek reprocessing, could accuse the United States of not having fulfilled its treaty obligations. This accusation, however ill-founded, could be made not only by the affected countries, but by any country opposed to the interests of the United States.

In the past, some individuals and groups have incorrectly asserted that the U.S. concerns with reprocessing of HEU spent nuclear fuel discussed above are inconsistent with the U.S. policy of continuing to grant prior consent[3] to Japan and Western European nations for reprocessing of power reactor spent nuclear fuel. The U.S. Government believes that the growing quantities of plutonium in international commerce do present a threat to the efforts of the United States and other countries to prevent the proliferation of nuclear weapons. In countries where material control and accounting or physical protection systems are not sufficiently rigorous, there is a risk of diversion or theft of such materials. In addition, even in countries with effective nuclear weapons nonproliferation commitments, the presence of unneeded stocks of plutonium could raise security concerns on the part of neighboring countries. Accordingly, the U.S. Government does not encourage the civil use of plutonium. Nevertheless, the United States is also committed to being a reliable nuclear trading partner and to avoiding interference in peaceful nuclear programs. Therefore, in Western Europe and Japan where there are well-established civil reprocessing and plutonium facilities and comprehensive nuclear weapons nonproliferation commitments, the United States will continue, in appropriate instances, to grant prior consent for reprocessing of plutonium-bearing spent fuels on a predictable and long-term basis. Undertaking the use of U.S. consent rights to block reprocessing would lead to confrontation with, and would jeopardize the support from, nations that are in accord with the broader U.S. nuclear weapons nonproliferation goals and agenda.

1.2 Purpose and Need For Agency Action

Curbing the spread of nuclear weapons has been an important foreign policy and national security objective of the United States for nearly half a century. The proposed action is one action among many being pursued by the United States to reduce the potential for the proliferation of nuclear weapons. More specifically, the proposed action is intended to support the U.S. policy objective of seeking to reduce, and eventually to eliminate, HEU from civil commerce.

The nuclear weapons proliferation concerns addressed by the proposed action stem from the use of HEU as fuel for foreign research reactors and the presence of large residual amounts of HEU in the spent nuclear fuel from these research reactors. HEU can be used directly in simple nuclear weapons. In the past, the United States has encouraged reductions in the use of HEU as research reactor fuel by conducting

---

3 *In general terms, the Atomic Energy Act of 1954, as amended, requires that before the United States may export nuclear material (e.g., enriched uranium), nuclear equipment, or sensitive nuclear technology to another country, that country must agree to obtain the consent (i.e., approval) of the United States before it may transfer that material, equipment, or technology to another country. However, in its 1988 agreement for cooperation with Japan, the United States has granted "prior consent" (i.e., approval in advance) for Japan to transfer power reactor spent fuel to France for reprocessing for the life of the agreement (the initial term is 30 years), subject to certain conditions relating to Japan's nuclear weapons nonproliferation activities. The U.S. Government has made a commitment to include a similar arrangement in new agreements that it is currently negotiating with Euratom and Switzerland.*

the RERTR program (to develop high-density LEU fuels to replace the HEU fuels). The United States also previously prevented the development of foreign stockpiles of HEU in foreign research reactor spent nuclear fuel by conducting the "Off-Site Fuels Policy" (i.e., by accepting the spent nuclear fuel into the United States and reprocessing it).

To illustrate the level of concern that exists regarding the proposed action, DOE has received letters from the U.S. Department of State, the Nuclear Regulatory Commission, the Arms Control and Disarmament Agency, the International Atomic Energy Agency, and the foreign research reactor operators, all urging DOE to resume acceptance and management of foreign research reactor spent nuclear fuel. Failure to manage this spent nuclear fuel would encourage international commerce in HEU.

HEU fuel in research reactors is more of a proliferation concern than the plutonium in these reactors for three reasons.

First, it is much easier to fashion a simple nuclear weapon out of HEU than plutonium. The chemical separation of HEU metal from HEU spent nuclear fuel would be simpler than the chemical separation of plutonium metal from LEU spent nuclear fuel. HEU is also less radioactive than plutonium, so it presents less of a health hazard to the people working with it. Furthermore, pure plutonium metal is extremely pyrophoric; that is, small chips of pure plutonium metal can ignite spontaneously unless the metal is handled very carefully in special facilities. All these problems with plutonium make HEU fuel a more attractive target for diversion into a nuclear weapons program.

Second, the amount of HEU that would be removed from current civil commerce under the proposed action is much greater than the amount of plutonium that would be produced in replacement LEU fuel elements in the same reactors. The proposed action involves the removal of approximately 4.6 metric tons (5.1 tons) of HEU. For comparison, the plutonium that would be produced in the replacement LEU nuclear fuel is so dilute that even if all the plutonium were somehow extracted, only about 120 kg (265 lb) of plutonium would be produced. This reduction in the amount of weapons grade material in circulation in the future would significantly reduce the threat of nuclear proliferation.

Third regarding fresh (not spent) nuclear fuel containing HEU, this nuclear material would be ideal for diversion into a nuclear weapons program because it would not require chemical separation as spent nuclear fuel would. In the absence of a policy to eliminate HEU from civil commerce, fresh HEU fuel would be shipped to the foreign research reactors from foreign suppliers such as Russia or China. Such shipments would present an additional proliferation risk that would not exist if research reactors worldwide operate on the LEU fuel cycle.

If the United States takes no action to accept foreign research reactor spent nuclear fuel, or otherwise eliminate much of the HEU it contains (e.g., by blending it with natural or depleted uranium to make LEU), one or more of the following events is almost certain to occur. First, some of the foreign research reactors would simply shut down. This does not solve the concerns regarding foreign research reactor spent nuclear fuel, however, since the countries whose reactors were forced to shut down could argue, rightly or wrongly, that the United States was not living up to its obligations under the *Treaty on the Non-Proliferation of Nuclear Weapons* to assist nonnuclear weapons States in the peaceful application of nuclear energy. In addition, the spent nuclear fuel already discharged by the foreign research reactors would still be in storage at the reactor sites. For about 70 percent of the foreign research reactors, this spent nuclear fuel would contain HEU that could be diverted into the production of nuclear weapons if it were removed from the spent nuclear fuel.

Second, some of the foreign research reactors might continue operating and allow their inventory of spent nuclear fuel to build up, much of it containing HEU. The United States could, theoretically, assist such countries (and those with shutdown reactors still holding spent nuclear fuel) in building modern, diversion-resistant spent nuclear fuel storage facilities. This could be extremely expensive, as there are approximately 104 research reactors located in 41 foreign countries. Furthermore, even if perfectly secure storage facilities were built, all that would be required to frustrate their function would be a coup or other change in government leaving a regime in power that is unconcerned about the proliferation of nuclear weapons. Then the spent nuclear fuel could be diverted into a weapons production program, despite the storage assistance that had been provided by the United States. It seems clear that the potential for such an event to occur would increase with the number of spent nuclear fuel stockpiles that are allowed to build up around the world and the length of time they exist.

Finally, some foreign research reactors would be likely to reprocess their spent nuclear fuel. Much of the U.S.-origin enriched uranium was exported under agreements that require prior consent by the United States before the spent nuclear fuel can be shipped to another country, as would be required for almost all reprocessing of foreign research reactor spent nuclear fuel containing uranium enriched in the United States. However, approximately 50 percent of the foreign research reactor spent nuclear fuel is located in countries where such prior consent is not required. The current practice of the most likely reprocessing plant (i.e., the facility in Dounreay, Scotland) is to allow the customer (i.e., the foreign research reactor operator) to specify the form of the separated uranium and its disposition. Thus, the foreign research reactor operator could specify that any separated HEU should be returned as HEU. Furthermore, neither Dounreay nor any other reprocessing facility currently accepts or has the capability to reprocess the high-density LEU fuels that the United States is encouraging foreign research reactors to use to replace the HEU fuels. Thus, in the absence of action to resolve the questions of the disposition of spent nuclear fuel, outlined above, any foreign research reactor operator that reprocesses to control the inventory of spent nuclear fuel must continue to use, or convert back to, fuel containing HEU. Reprocessing under these circumstances leads to perpetuation of the HEU fuel cycle.

While there is some danger of diversion of the HEU in the spent nuclear fuel while it is in storage, the threat is relatively low since the uranium is an integral part of the solid metal spent nuclear fuel elements and is mixed with highly radioactive fission products. A sophisticated chemical processing plant would be required to separate the uranium and convert it into a form suitable for use in a nuclear weapon. However, once the uranium has been separated from the spent nuclear fuel in a reprocessing plant, the situation is fundamentally changed. Any HEU separated in a reprocessing plant would be readily usable for nuclear weapons production, essentially the same as fresh HEU that has never been in a reactor. Therefore, despite U.S. confidence in the capability and determination of the United Kingdom and France to properly safeguard any separated HEU in their reprocessing plants, once the uranium leaves their facilities, the potential for illegal diversion of the material during transit or in the country of destination increases markedly. The rate of increase for potential diversion depends both on the countries and international waterways through which the material must be shipped, the country of final destination, and the number and size of the shipments being made.

Although it is unlikely that all the HEU that has been exported can be recovered or blended down to LEU through the proposed action, it is in the best interest of the United States to make every effort to ensure the proper management of the largest fraction of this material. Arrangements would have to be worked out with foreign reprocessors that would be supportive of U.S. nuclear weapons nonproliferation objectives to minimize the civil use of HEU worldwide.

By proposing a policy for management of certain foreign research reactor spent nuclear fuel, DOE and the Department of State do not seek to indefinitely accept or otherwise manage spent nuclear fuel from foreign research reactors. Rather, the purpose of the proposed new policy is to remove as much U.S.-origin HEU as possible from international commerce while giving the foreign research reactors and their host countries time to convert to operation with LEU fuel and make their own arrangements for disposition of subsequently generated LEU spent nuclear fuel. The foreign research reactor operators and countries in which the research reactors are operating must be prepared to implement their own arrangements for disposition of their spent nuclear fuel after the policy expires.

1.3 Scope of the Environmental Impact Statement (EIS)

This EIS evaluates the potential environmental impacts that could result from the DOE and Department of State joint proposal to adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel containing uranium enriched in the United States would be covered by the proposed policy. The purpose of the proposed policy is to promote U.S. nuclear weapons nonproliferation policy objectives, specifically by seeking to reduce, and ultimately eliminating, HEU from civilian commerce. This EIS identifies and evaluates the potential environmental impacts of management alternatives to the proposed action. Implementation of Management Alternative 1 to the proposed action could include the receipt of foreign research reactor spent nuclear fuel at one or more U.S. marine ports of entry, overland transport to one or more DOE sites, and management (interim storage and ultimate disposition) in the United States. DOE will also analyze near-term chemical separation as an alternative to storing the intact spent nuclear fuel pending ultimate disposition.

Under Management Alternative 1 to the proposed action, DOE would accept or otherwise manage spent nuclear fuel containing HEU and LEU from 41 foreign nations, if such spent nuclear fuel is already discharged[4] or would be discharged within the 10-year policy period. Figure 1-1 displays the geographic locations of these nations. The United States would bear the full cost for the management of the foreign research reactor spent nuclear fuel from developing nations.[5] For developed nations, however, the United States would charge a fee for spent nuclear fuel management activities conducted by the United States.[6]

DOE recognizes that Figure 1-1 lists nations that would currently not be considered to be nuclear weapons proliferation risks. History indicates that the United States cannot predict today, with assurance, which countries may develop into proliferation risks in the future. On the other hand, there are good reasons for accepting spent nuclear fuel from nations that are in accord with U.S. nuclear nonproliferation goals, and that have stable governments and excellent nonproliferation credentials. First, several of these countries manufacture research reactors for sale to third world countries. If the United States refuses to help these countries with the management of the spent nuclear fuel from their research reactors, several of their reactors are likely to convert to use of HEU for fuel. If that occurs, their customers in the third world countries would probably also demand to be supplied with reactors fueled with HEU.

Second, both the Soviet Union and China also exported research reactors in the past. The United States is currently engaged in discussions to convince Russia (as the successor supplier to the Soviet Union's allies) and China to work with their nuclear fuel customers to convert their research reactors from HEU to LEU.

4 *"Discharged" refers to removal of irradiated fuel from a reactor.*

5 *Developing nations are defined by the World Development Report as having other than high-income economies (World Bank, 1994).*

6 *For purposes of determining which nations are eligible for assistance from the United States in handling their foreign research reactor spent nuclear fuel, Taiwan is considered to be in the high-income economy category.*

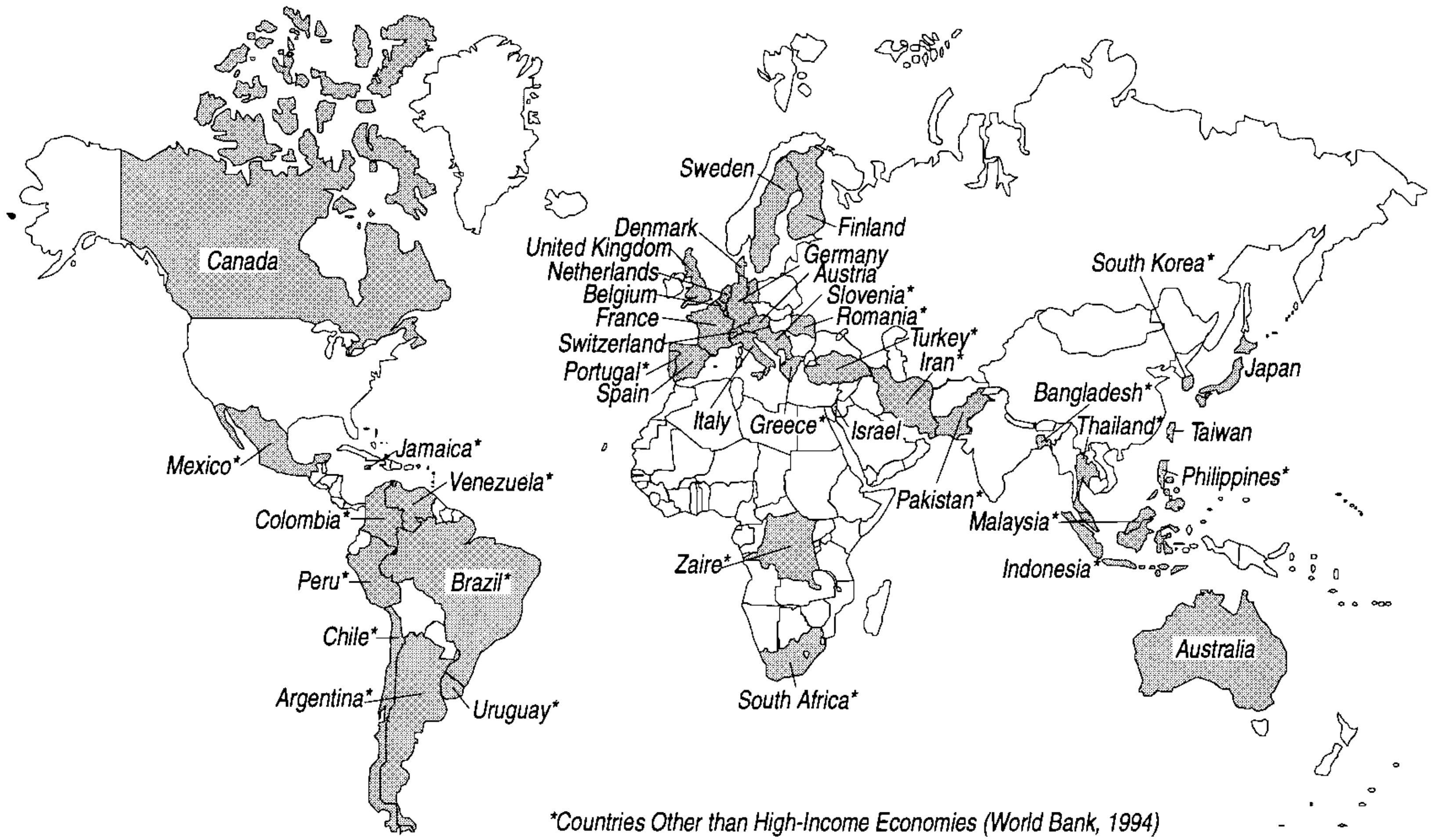


Figure 1-1 Nations with Research Reactors that are Holding or are Expected to Generate Spent Nuclear Fuel Containing Uranium Enriched in the United States

If the United States were to take action that caused its nuclear fuel customers to reconvert their research reactors to HEU, that would likely ensure the failure of U.S. efforts to get Russia and China to encourage their nuclear fuel customers to switch to LEU.

A second alternative (Management Alternative 2) for implementation of the proposed action has been identified and evaluated in this EIS. This alternative includes two subalternatives: (1) to provide assistance to foreign nations with storage of their spent nuclear fuel overseas, and (2) to provide non-technical assistance (financial and/or logistical) in the reprocessing of their spent nuclear fuel. The third alternative (Management Alternative 3) would consist of a combination of various elements of Management Alternatives 1 and 2. For example, a portion of the spent nuclear fuel could be managed overseas, and the remaining portion could be managed in the United States. A No Action Alternative is also evaluated in this EIS.

Under any of these action alternatives, no definitive proposals can be specified at this time for management of the foreign research reactor spent nuclear fuel beyond a 40-year interim management period because insufficient data are available to allow future management proposals to be defined, or for the potential environmental impacts of the final disposition of spent nuclear fuel to be evaluated. As a result, the EIS analysis for the time period beyond 40 years is qualitative rather than quantitative. The qualitative assessment includes consideration of disposal of intact foreign research reactor spent nuclear fuel, as well as disposal of vitrified high-level waste resulting from chemical separation of foreign research reactor spent nuclear fuel.

Certain potential actions discussed in this EIS will depend on decisions to be made under other National Environmental Policy Act (NEPA) analyses. Specifically, the site at which the foreign research reactor spent nuclear fuel would be managed, if accepted in the United States, will be selected based on the analyses documented in the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (Programmatic SNF&INEL Final EIS) (DOE, 1995c). An exception to this would occur if the "No Action" alternative, or any other alternative that does not include acceptance of foreign research reactor spent nuclear fuel, were selected after completion of the Programmatic SNF&INEL Final EIS. In that case, any site at which foreign research reactor spent nuclear fuel management activities might be conducted in the United States would be selected pursuant to the analyses in this EIS.

The Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995c) was issued on May 30, 1995. In accordance with this Record of Decision, all of the aluminum clad foreign research reactor spent nuclear fuel accepted by DOE would be managed at the Savannah River Site in South Carolina, and any other foreign research reactor spent nuclear fuel to be accepted by DOE would be managed at the Idaho National Engineering Laboratory. This is why the Comment Response Document (Volume 3) focuses on the Savannah River Site and Idaho National Engineering Laboratory as sites where any spent fuel accepted in the United States under the proposed policy would be managed, consistent with the Programmatic SNF&INEL Final EIS Record of Decision. Nevertheless, all five of the spent nuclear fuel management sites originally considered in the Draft EIS have been kept in this Final EIS to maintain maximum consistency with the analyses provided in the Programmatic SNF&INEL EIS (DOE, 1995c and 1994m).

In this EIS, DOE and the Department of State, in consultation with other government agencies, designate the acceptance and management of foreign research reactor spent nuclear fuel and target material in the United States as the preferred alternative.

1.4 Decisions to be Made Based on this EIS

The principal policy decision for which this EIS will provide a basis is whether DOE and the Department of State should adopt a policy for the United States to manage foreign research reactor spent nuclear fuel containing U.S.-enriched uranium. A necessary part of this decision is the amount and types of foreign research reactor spent nuclear fuel to be managed under such a policy.

If a decision is made to adopt such a policy, then decisions on management implementation must also be made. This EIS is intended to provide the necessary analysis, not only for the decision on adoption of a policy, but also for decisions on management implementation of such a policy, if adopted. Should the decision be to manage the spent nuclear fuel in the United States, decisions to be made on implementation of such a policy include:

- the duration of any foreign research reactor spent nuclear fuel management policy;
- the modes of ocean and overland transport required for shipping any foreign research reactor spent nuclear fuel to be accepted into the United States;
- the ports of entry through which DOE would receive foreign research reactor spent nuclear fuel;
- the need for the construction of new facilities or modification of existing facilities to manage the foreign research reactor spent nuclear fuel to be accepted, and the design, construction, and operation of any such new or modified facilities;
- whether to use near-term chemical separation of the foreign research reactor spent nuclear fuel, as an alternative to storing intact foreign research reactor spent nuclear fuel; and
- whether to accept uranium target material that was enriched in the United States and irradiated in foreign research reactors during the production of molybdenum-99 for medical purposes (in addition to the foreign research reactor spent nuclear fuel discussed above).

1.5 Relationship of this EIS to Other NEPA Documents and Reports Relating to Spent Nuclear Fuel Management

The relationship of this EIS to other DOE NEPA reviews, either completed or currently under preparation, and other DOE analyses is discussed in this section. These reviews and analyses are:

1. *Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel.* This Environmental Assessment covers marine transport, receipt, overland transport, and interim wet storage in the Receiving Basin for Offsite Fuels (RBOF) at the Savannah River Site of up to 409 elements of spent nuclear fuel from foreign research reactors. The Environmental Assessment and associated Finding of No Significant Impact were issued on April 22, 1994.

   The proposed action analyzed in this Environmental Assessment was intended to ensure that the eight research reactors from which urgent-relief spent nuclear fuel shipments were proposed would continue to participate in the RERTR program (a key U.S. nuclear weapons nonproliferation program) until this EIS could be completed and a decision made on whether to adopt and implement the proposed policy to manage foreign research reactor spent nuclear fuel containing uranium enriched in the United States.

2. *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement (Programmatic SNF&INEL EIS).* Volume 1 analyzes at a programmatic level the potential environmental impacts over the next 40 years of alternatives related to the transportation, receipt, processing, and storage of spent nuclear fuel under the responsibility of DOE. This EIS was prepared in compliance with the order of the U.S. District Court for the District of Idaho [*Public Service Company of Colorado v. Andrus*, Memorandum of Opinion (December 22, 1993)]. The Final EIS and the Record of Decision were published on April 28, 1995 and May 30, 1995, respectively. This EIS formed the basis for deciding, on a programmatic level, which sites will be used for the management of the various types of spent nuclear fuel to which DOE holds title. It included the amount of foreign research reactor spent nuclear fuel that might be accepted in its assessment of potential impacts, and addressed the sites at which the foreign research reactor spent nuclear fuel could be stored if a decision is made to accept foreign research reactor spent nuclear fuel. The Record of Decision indicated that aluminum clad spent fuel will be consolidated at the Savannah River Site and non-aluminum clad fuel will be managed at the Idaho National Engineering Laboratory. On October 17, 1995, litigation with the State of Idaho was settled by stipulation of the parties and entry of a Consent Order. This settlement would provide for the transportation of up to 61 shipments of foreign research reactor spent fuel to Idaho National Engineering Laboratory prior to the year 2000, if DOE and the Department of State choose to adopt a policy of accepting such foreign research reactor spent nuclear fuel. After the year 2000, additional shipments of such spent nuclear fuel could be made to Idaho National Engineering Laboratory under the stipulated settlement and Consent Order. Notwithstanding the Record of Decision, a full analysis of all five management sites considered in the Programmatic SNF&INEL Final EIS has also been included in this EIS to maintain maximum consistency with the analysis provided in the Programmatic SNF&INEL EIS.

3. *Waste Management Programmatic EIS.* The Waste Management Programmatic EIS has evolved from the formerly named Environmental Management Programmatic EIS, as was most recently described in a *Federal Register* Notice (55 FR 42633), on August 22, 1990. The Draft Waste Management Programmatic EIS was issued in August 1995 and analyzes programmatic alternatives for DOE-wide management of five waste types: high-level radioactive waste, low-level radioactive waste, low-level mixed waste, transuranic waste, and hazardous waste. DOE expects to issue the Final Waste Management Programmatic EIS in late 1996.

4. *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.* The Spent Fuel Working Group Report, dated November 1993, presents a comprehensive assessment of the conditions of spent nuclear fuel and other irradiated materials stored at DOE facilities. Eight DOE sites were identified as containing storage facilities with major vulnerabilities that need to be resolved. The vulnerabilities identified in this report have been considered in the analysis of the actions that would be involved in management of foreign research reactor spent nuclear fuel in the United States.

5. *Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities.* The Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities is a three-phased approach to remedy vulnerabilities identified in the Spent Fuel Working Group Report. The Phase-I Plan of Action, dated

February 1994, addressed 31 of 33 high-priority vulnerabilities and 48 lower-priority issues. The Phase-II Plan of Action, dated April 1994, was the product of follow-on work to the Phase-I report, and resolved a majority of the funding issues associated with spent fuel vulnerabilities. The Phase-III Plan of Action, issued in October 1994, focused on the resolution of critical policy issues and provided individual action plans that addressed all the identified vulnerabilities. In the preparation of this EIS on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, the actions being taken to resolve these issues have been evaluated to ascertain that any existing facilities considered for use in receipt, handling, or storage of foreign research reactor spent nuclear fuel are capable of performing satisfactorily. The EIS evaluates storage capabilities and alternatives at the interim storage sites and whether potential storage sites would be capable of immediately implementing the proposed action.

6. *Interim Management of Nuclear Materials Final Environmental Impact Statement.* This EIS considers the impacts of managing nuclear materials stored at the Savannah River Site, including stabilization by separating the nuclear materials from fission and decay products in the chemical separation facilities and conversion of the resulting liquids to solids in waste and material conversion facilities, including the Defense Waste Processing Facility, the FB-, HB, and FA-Lines, and the saltstone facility. The Final EIS was issued in October 1995. A Record of Decision and Notice of Preferred Alternative was published in the *Federal Register* (60 FR 65300) on December 19, 1995. Decisions were made for the majority of materials covered by the EIS in the Record of Decision and processing Mark-16 and Mark-22 fuels and blending down the resulting HEU to LEU was identified as the preferred alternative for those materials. These fuels are similar to the aluminum-based foreign research reactor spent nuclear fuel although significant corrosion has been identified. An amended Record of Decision is expected soon regarding the Mark-16 and Mark-22 fuels. DOE has taken and will take into consideration all Records of Decision on the Interim Management of Nuclear Materials Final EIS in the preparation of this EIS and in reaching a decision on how to implement the proposed policy, if adopted.

7. *Disposition of Surplus Highly Enriched Uranium Environmental Impact Statement.* The Draft EIS, issued in October 1995, assesses the environmental impacts of alternatives for the disposition of U.S.-origin HEU that has or may be declared surplus to national defense and defense-related program needs, in order to eliminate the nuclear proliferation risk and, where practical, recover economic value and peaceful, beneficial reuse of the material. Under the preferred alternative, the HEU would be blended to LEU at four sites, including the Savannah River Site; under this alternative, most of the HEU would be blended to LEU as fuel feed for commercial nuclear power plants to generate electricity, while that which cannot meet commercial fuel specifications would be blended to low level waste. The Final EIS and Record of Decision are scheduled for April and May 1996, respectively.

8. *Storage and Disposition of Weapons-Usable Fissile Materials Programmatic Environmental Impact Statement.* This EIS will assess the environmental impacts of reasonable alternatives for safe, secure and internationally-accountable long-term storage of non-surplus, U.S.-origin plutonium and HEU, long-term storage of plutonium and HEU that are not part of the strategic reserves need for weapons research and development, and post-interim storage of surplus fissile material prior to disposition. This programmatic EIS will also analyze reasonable alternative strategies and technologies for disposition of U.S.-origin plutonium that is or may be declared surplus to defense and defense-related program needs,

in order to eliminate the proliferation risk by making the material as proliferation-resistant as spent fuel. The Savannah River Site will be analyzed in conjunction with the various alternatives for both storage and disposition. The Notice of Intent to prepare a programmatic EIS was issued in June 1994 (59 FR 31985), and the scope of the EIS was revised in April 1995 (60 FR 17344). The Draft EIS is expected in early 1996, and the Final EIS and Record of Decision are expected in late 1996.

9. *Environmental Impact Statement Evaluating Container Systems for the Management of Spent Nuclear Fuel.* This EIS was originally titled *Environmental Impact Statement for a Multi-Purpose Canister System for the Management of Civilian and Naval Spent Nuclear Fuel.* This EIS, as described in 59 FR 53442 (1994), was intended to address the potential environmental impacts associated with alternative systems for storage and transport of spent nuclear fuel assemblies for civilian spent nuclear fuel. DOE decided for programmatic reasons in November 1995 to withdraw its proposal to prepare this EIS. The Department of the Navy has announced in a *Federal Register* Notice (60 FR 62829) that it will take the lead in preparing this EIS for evaluating container systems for the management of Navy spent nuclear fuel. DOE is a cooperative agency on this EIS.

10. *F-Canyon Plutonium Solutions Environmental Impact Statement.* This EIS evaluated the potential environmental impacts over the next 10 years of alternatives for stabilization of plutonium solutions currently stored in the F-Canyon at the Savannah River Site. The plutonium solutions remain from reprocessing operations that DOE suspended in 1992 at the Savannah River Site. The Record of Decision for this EIS announced that DOE would implement the preferred alternative analyzed in the EIS. This alternative is to process the plutonium solutions to plutonium metal.

11. *Defense Waste Processing Facility Supplemental Environmental Impact Statement.* This Supplemental EIS examines the cumulative environmental impacts of modifications made to the Defense Waste Processing Facility and associated high-level waste facilities at the Savannah River Site since the issuance of the 1982 EIS. The preferred alternative for the proposed action under this Supplemental EIS is to continue construction and begin operation of the Defense Waste Processing Facility as designed. The Final Supplemental EIS was completed in November 1994 and the Record of Decision was issued on April 12, 1995 (60 FR 18589). The DOE decision was to complete construction and startup testing and begin operation of the facility as currently designed. One of the Implementation Alternatives considered in the Foreign Research Reactor Spent Nuclear Fuel EIS is to chemically separate a portion of the foreign research reactor spent nuclear fuel at the Savannah River Site and vitrify the resulting high-level radioactive waste in the Defense Waste Processing Facility.

12. *Tritium Supply and Recycling Programmatic Environmental Impact Statement.* This Programmatic EIS evaluated the siting, construction and operation of tritium supply technology alternatives and the recycling facilities at five candidate DOE sites. The EIS also evaluated the use of a commercial reactor for producing tritium. Currently, DOE does not have the capability to produce tritium in the required amounts. The Savannah River Site in South Carolina, which will receive the aluminum-based foreign research reactor spent nuclear fuel, has been identified by DOE as the preferred site for an accelerator, should one be constructed, and the site for the upgrade and consolidation of existing recycling facilities.

The Final Programmatic EIS was completed and issued to the public in October 1995. The DOE Record of Decision was issued on December 12, 1995 (60 FR 63891) with a decision to implement the preferred alternatives.

13. *Stockpile Stewardship and Management Programmatic Environmental Impact Statement.* This EIS was originally a part of the Nuclear Weapons Complex Reconfiguration Programmatic Environmental Impact Statement. The Notice of Intent of this EIS was published on June 14, 1995 (60 FR 31291) after a prescoping workshop on May 19, 1995. This Programmatic EIS will examine activities required to maintain a high level of confidence in the safety and reliability of a reduced stockpile of nuclear weapons in the absence of underground nuclear testing. The Savannah River Site at Aiken, South Carolina, which houses the tritium loading/unloading and surveillance of tritium reservoirs, will receive the aluminum-based foreign research reactor spent nuclear fuel should the proposed action be implemented in the United States. This draft EIS is expected to be issued for public comment in February 1996.

1.6 Structure of this EIS

The remainder of this EIS is structured as follows:

- Chapter 2 presents the proposed action, describes management alternatives for implementation of the proposed action, alternative means of implementing each management alternative, and a No Action Alternative. Chapter 2 specifies the Preferred Alternative that has been developed by DOE and the Department of State.
- Chapter 3 characterizes the affected environments at potential ports of entry and at potential foreign research reactor spent nuclear fuel management locations.
- Chapter 4 addresses the policy considerations and the potential environmental impacts of each management alternative for implementation of the proposed action, alternative means of implementing each management alternative, and a No Action Alternative.
- Chapter 5 describes the international and domestic regulations governing radioactive materials that apply to DOE actions that might be taken under this EIS.
- Chapters 6, 7, and 8 contain primarily reference information, such as the List of Preparers, Agencies Consulted, and References, respectively.

The appendices to this document present details of the evaluations and analyses performed for this EIS.

2. Proposed Action and Alternatives

This chapter states the proposed action and describes the management alternatives analyzed in this Environmental Impact Statement (EIS) for implementation of the proposed action. Environmental and policy impacts from the management alternatives are presented in Chapter 4.

2.1 Overview of the Proposed Action and Alternatives

The U.S. Department of Energy (DOE) and Department of State are jointly proposing to adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel containing uranium enriched in the United States would be covered by the proposed action. The purpose of the proposed action is to promote U.S. nuclear weapons nonproliferation policy objectives, specifically by seeking to reduce, and eventually eliminate, highly-enriched uranium (HEU) from civilian commerce.

To implement the proposed action, DOE and the Department of State have considered three foreign research reactor spent nuclear fuel management alternatives. They are:

1. To accept and manage foreign research reactor spent nuclear fuel in the United States (Management Alternative 1);
2. To facilitate the management of foreign research reactor spent nuclear fuel at one or more foreign locations (Management Alternative 2); and
3. A combination of elements from Management Alternatives 1 and 2 (Management Alternative 3, Hybrid Alternative).

The management alternatives of the proposed action are portrayed in Figure 2-1 and are discussed in more detail in Sections 2.2, 2.3, and 2.4.

A No Action Alternative on the part of DOE and the Department of State to address the status of the foreign research reactor spent nuclear fuel has also been considered in this EIS. The No Action Alternative is discussed in Section 2.5.

DOE and the Department of State have identified a preferred alternative for the proposed action. The preferred alternative is described in Section 2.9.

The foundation for the analysis presented in Chapter 4 of this EIS is the evaluation of the components that comprise the basic implementation of Management Alternative 1. The basic implementation concept is an attempt by DOE and the Department of State to avoid unnecessary repetition by selecting a reasonable option for each component and examining them in detail under Management Alternative 1. Since the No Action Alternative would not have any direct environmental impacts in the United States, it requires only policy analysis in this EIS. Management Alternatives 1, 2, and 3, however, would all have environmental impacts in the United States, and the components of the basic implementation provide the parameters with which to analyze their potential environmental impacts in this EIS.

The detail of analysis provided for the basic implementation components is based on the fact that some variation of these components is utilized in each implementation alternative under Management Alternative 1, as well as in Management Alternative 3. In this way, analysis of the implementation

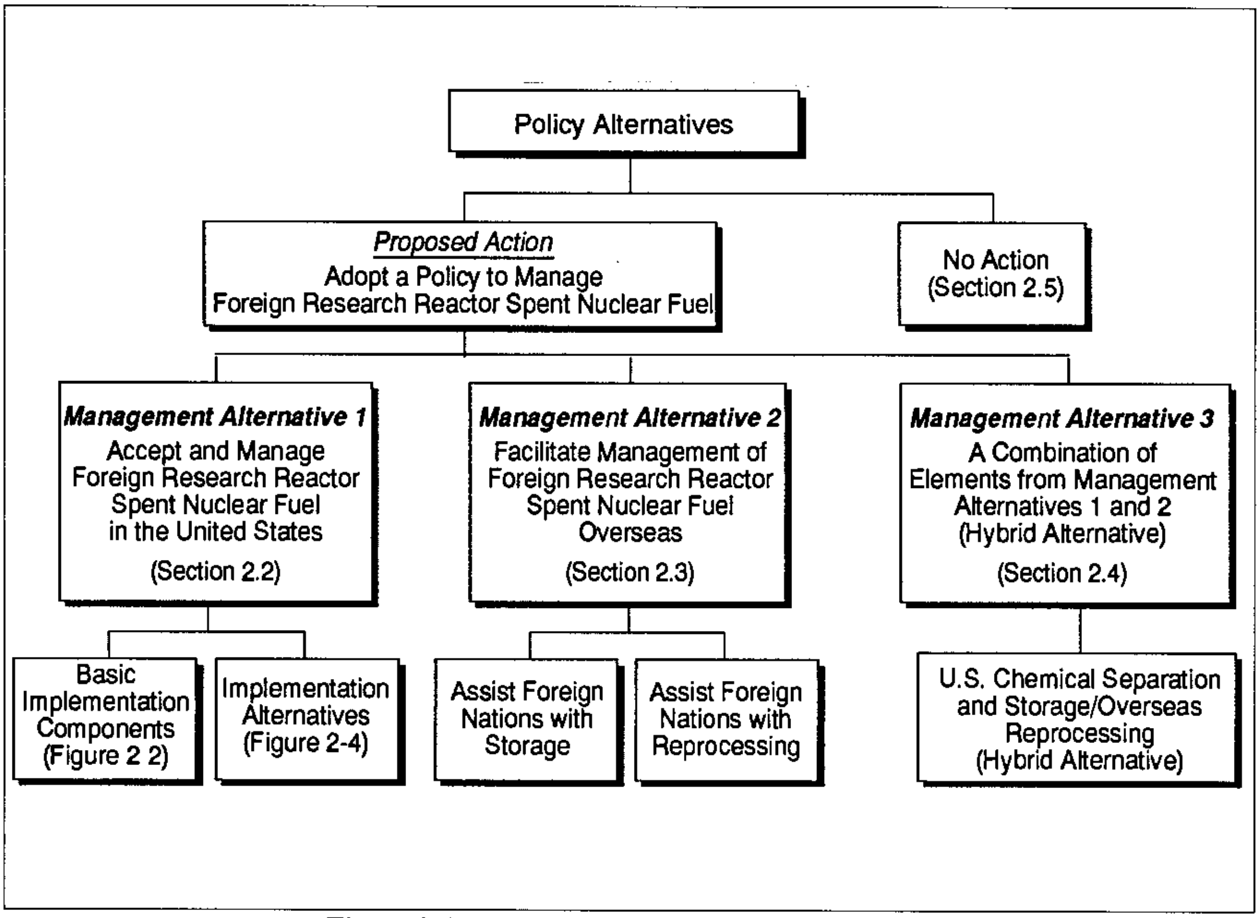


Figure 2-1 Policy and Management Alternatives

alternatives, as well as Management Alternative 3, can be tiered from the analysis of the basic implementation. In and of itself, the basic implementation of Management Alternative 1 is a viable implementation alternative for consideration under Management Alternative 1, along with the other implementation alternatives discussed below. However, the level of detail contained in the analysis of the basic implementation does not indicate any preference for this alternative. Rather, it merely eliminates the need to duplicate information later in the analysis.

The components of the basic implementation of Management Alternative 1 would consist of the following:

1. A policy duration of 10 years.

2. A financing arrangement by which the United States would bear the full cost for transporting and managing the foreign research reactor spent nuclear fuel received from developing countries, but would charge developed countries a competitive fee.

3. The receipt of a fixed amount of foreign research reactor spent nuclear fuel containing uranium enriched in the United States. This fixed amount is up to approximately 22,700 foreign research reactor spent nuclear fuel elements and is based on estimated inventories of foreign research reactor spent nuclear fuel currently stored or to be generated in the 10-year policy period.

4. Taking title to the foreign research reactor spent nuclear fuel at the U.S. territorial waters limit (19 km or 12 mi), or continental U.S. borders for shipments from Canada.

5. Marine transport of the foreign research reactor spent nuclear fuel by chartered and/or regularly scheduled commercial ships.

6. Ports of entry that qualify on the basis of criteria discussed in this EIS.

7. Ground transport from ports of entry to storage sites, and between sites (by truck, rail, or barge, or a combination of these modes.)

8. Potential storage sites identified in the Programmatic SNF&INEL Final EIS (DOE, 1995c) for foreign research reactor spent nuclear fuel, namely the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

9. Use of dry storage technology for construction of new storage facilities.

The basic implementation components are depicted in Figure 2-2 and described in Section 2.2.1. Environmental impacts and policy considerations of the basic implementation components of Management Alternative 1 are presented in Section 4.2.

Utilizing the components provided above, DOE has evaluated seven implementation alternatives for Management Alternative 1 in addition to the basic implementation. Each implementation alternative is comprised of the same components as the basic implementation; however, for the purpose of analysis, one of the components has been varied. The seven implementation alternatives are given below.

1. Acceptance of amounts of foreign research reactor spent nuclear fuel different from the amounts identified in the basic implementation.

2. Acceptance of foreign research reactor spent nuclear fuel for a period of time different from the time period identified in the basic implementation.

3. Financial arrangements different from those identified in the basic implementation.

4. Taking title to foreign research reactor spent nuclear fuel at locations different from those identified in the basic implementation.

5. Use of wet storage technology for construction of new storage facilities instead of dry storage technology as identified in the basic implementation.

6. Use of near term conventional chemical separation in the United States to reduce the duration of, and amount of, spent nuclear fuel storage required.

7. Use of new developmental treatment and/or packaging technologies in addition to storage as identified in the basic implementation.

Implementation alternatives for Management Alternative 1 are discussed in Section 2.2.2. The environmental impacts and policy considerations of the implementation alternatives are discussed in Section 4.3.

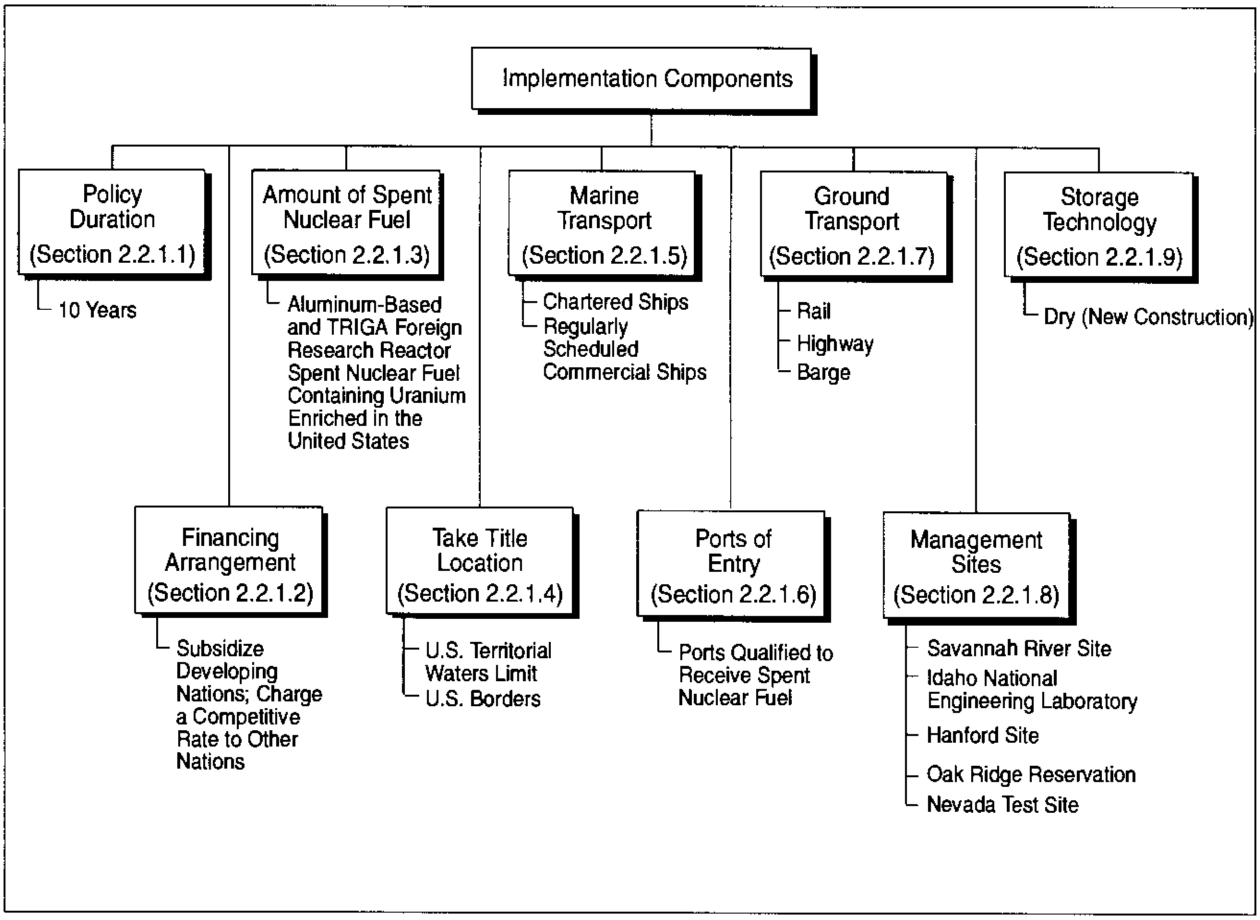


Figure 2-2 Basic Implementation Components

Qualifying Fuel Types and Policy Stipulations

This policy applies solely to aluminum-based and TRIGA[1] research reactor fuels and target materials containing HEU and low enriched uranium (LEU) of U.S. origin. Aluminum-based fuel is clad in aluminum and has an active fuel region that consists of an alloy of uranium and aluminum or a dispersion of uranium-aluminide, uranium-oxide[2] or uranium-silicide in aluminum. TRIGA fuel consists of an alloy of uranium and zirconium and is clad in either aluminum or stainless steel. Fuels containing significant quantities of Uranium-233 (^{233}U) are excluded. Target materials are the residual materials from isotope production targets in research reactors.

The policy would include the following stipulations:

- Spent nuclear fuel (either and/or both HEU and LEU) would be accepted from research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective.

1 *TRIGA stands for Training, Research, Isotope, General Atomic reactors.*
2 *This uranium-oxide composition refers to aluminum-clad fuel plates or tubes containing dispersions of U_3O_8 in aluminum. It does not include fuels containing UO_2 pellets clad in aluminum, zircaloy, stainless steel, or other materials, or uranium-silicide in aluminum.*

- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors which operate on HEU fuel when the policy becomes effective and which agree to convert to LEU fuel. Spent nuclear fuel would not be accepted from research reactors that could convert to LEU fuel but refuse to do so.
- Spent nuclear fuel (HEU) would be accepted from research reactors having lifetime cores, from research reactors planning to shut down by a specific date while the policy is in effect, and from research reactors for which a suitable LEU fuel is not available.
- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors that are already shut down.
- Unirradiated fuel (HEU and/or LEU) from eligible research reactors would be accepted as spent nuclear fuel.
- For research reactors with both HEU and LEU spent nuclear fuel available for shipment, LEU spent nuclear fuel would not be accepted until the HEU spent nuclear fuel is exhausted, unless there are extenuating circumstances (e.g., deterioration of one or more LEU elements sufficient to cause a safety problem).
- Spent nuclear fuel (HEU and/or LEU) would not be accepted from new research reactors starting operation after the date of implementation of the policy.

Ultimate Disposition

Ultimate disposition of DOE's spent nuclear fuel, including foreign research reactor spent nuclear fuel, is a high priority. For planning purposes, DOE has determined that its spent nuclear fuel that is not otherwise managed (e.g., chemically separated, with the high-level waste being converted into a vitrified glass for repository disposal) is authorized for disposal in a geologic repository. The Nuclear Waste Policy Act of 1982 (as amended) authorizes disposal of the foreign research reactor spent nuclear fuel in a geologic repository (if DOE takes title to such spent nuclear fuel). However, since the repository characterization program is in its early stages, the waste acceptance criteria for deposit of DOE's spent nuclear fuel in a repository have not been developed. Thus, a determination cannot be made at this time as to the requirements that must be met to allow placement of the foreign research reactor spent nuclear fuel in the repository. As a result, the EIS analysis for the time period beyond 40 years is qualitative rather than quantitative. The qualitative assessment includes consideration of disposal of intact foreign research reactor spent nuclear fuel, disposal of vitrified high-level waste resulting from chemical separation, as well as utilization of various potential new technologies to process the spent nuclear fuel into a more stable form prior to its ultimate disposition. In the event that the geologic repository schedule is delayed beyond the 40-year program period, DOE would continue to manage the foreign research reactor spent nuclear fuel or any resultant stable waste forms in existing facilities at the DOE management site(s) until a geologic repository becomes available. Decisions regarding the actual disposition of DOE's spent nuclear fuel will follow appropriate review under the National Environmental Policy Act (NEPA).

2.2 Management Alternative 1 - Manage Foreign Research Reactor Spent Nuclear Fuel in the United States

2.2.1 Basic Implementation Components

2.2.1.1 Policy Duration

The policy duration of the basic implementation of Management Alternative 1 would be 10 years, beginning on the date when the management policy becomes effective. Spent nuclear fuel containing HEU and LEU of U.S. origin that is currently being stored or is to be generated during the 10-year period of the policy would be accepted.

Actual shipments of spent nuclear fuel to the United States could be made for a period of 13 years starting from the effective date of the policy implementation, as long as the spent nuclear fuel was generated within the 10-year policy period. The 3 additional years would allow for a cooling time for fuel discharged from a reactor late in the policy period, logistics in arranging for shipment of this fuel, as well as other unplanned for potential delays.

2.2.1.2 Financing Arrangements

The United States would bear the full cost of transporting and managing the foreign research reactor spent nuclear fuel received from developing countries. Developing countries are defined by the World Bank as those countries having other than high-income economies (World Bank, 1994). For developed countries, however, the United States would charge a competitive fee for the handling, storage, conditioning (as needed), and any disposal activities conducted by the United States. Tables 2-1 and 2-2, which provide estimates of the number of elements that may be accepted, identify those countries defined as developing countries.

2.2.1.3 Amount of Foreign Research Reactor Spent Nuclear Fuel

The analysis in this EIS is based primarily on the number of individual elements of foreign research reactor spent nuclear fuel that could be accepted. When appropriate, the analysis also uses two other measures to express the amounts in understandable terms:

- Mass of Heavy Metal. This is the mass of all the heavy metal atoms in the spent nuclear fuel (mostly uranium), excluding the mass of other materials such as alloys, cladding, and structural materials. The international standard unit of measure for this quantity is metric tons of heavy metal (MTHM).
- Volume. The volume of the spent nuclear fuel is important because it determines the number of shipments and the storage space required. The volume is expressed in cubic meters.

The amount of foreign research reactor spent nuclear fuel that would be accepted under the basic implementation is up to approximately 19.2 MTHM, with a volume of approximately 110 m^3 (4,100 ft^3), representing approximately 22,700 individual spent nuclear fuel elements. The number of elements cited for acceptance under the policy includes those elements at issue in the Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994m).

Table 2-1 Estimated Number of Aluminum-Based Spent Nuclear Fuel Elements Generated by Foreign Research Reactor Operators by January 2006

Country	*Estimated Number of Spent Nuclear Fuel Elements*	*Initial Mass of Uranium (MTHM)*[a]	*Estimated Number of Shipments*
Argentina[b]	283	0.071	9
Australia	975	0.427	9
Austria	157	0.191	5
Belgium	1,766	0.730	59
Brazil[b]	155	0.099	5
Canada	2,831	4.478	116
Chile[b]	58	0.012	2
Colombia[b]	16	0.002	1
Denmark	660	0.529	22
France	1,962	3.442	149
Germany	1,504	0.909	49
Greece[b]	239	0.113	8
Indonesia[b]	198	0.236	6
Iran[b]	29	0.006	1
Israel	192	0.111	6
Italy	150	0.043	5
Jamaica[b]	2	0.001	1
Japan	2,981	3.134	99
Korea (South)[b]	168	0.321	7
Netherlands	1,488	1.404	49
Pakistan[b]	82	0.016	3
Peru[b]	29	0.039	1
Philippines[b]	50	0.024	2
Portugal[b]	88	0.054	3
South Africa[b]	50	0.010	2
Spain (from Scotland)[c]	40	0.016	1
Sweden	1,113	1.374	37
Switzerland	159	0.128	5
Taiwan	127	0.066	4
Thailand[b]	31	0.005	1
Turkey[b]	69	0.089	2
United Kingdom	12	0.004	1
Uruguay[b]	19	0.018	1
Venezuela[b]	120	0.082	4
Total	17,803	**18.184**	**675**

[a] *To derive uranium mass in kilograms, multiply the amounts by 1,000.*

[b] *Countries with other than high-income economies (World Bank, 1994).*

[c] *40 Spent nuclear fuel elements of Spain's JEN-1 Reactor core are stored in Dounreay, Scotland.*

Implementation of Management Alternative 1 would involve less than 1 percent of the total mass of heavy metal that DOE currently manages as spent nuclear fuel (DOE, 1994c), and approximately 10 percent of the volume.

Table 2-2 Estimated Number of TRIGA[a] Reactor Spent Nuclear Fuel Elements Generated by Foreign Research Reactor Operators by January 2006

Country	Estimated Number of Spent Nuclear Fuel Elements	Initial Mass of Uranium (MTHM)[b]	Estimated Number of Shipments
Austria	106	0.020	3
Bangladesh[c]	100	0.049	3
Brazil[c]	75	0.014	3
Finland	171	0.033	6
Germany	358	0.068	12
Indonesia[c]	245	0.047	8
Italy	386	0.072	13
Japan	326	0.062	11
Korea (South)[c]	336	0.064	11
Malaysia[c]	94	0.047	3
Mexico[c]	186	0.035	6
Philippines[c]	128	0.079	4
Romania[c]	1,451	0.189	48
Slovenia[c]	393	0.075	13
Taiwan	144	0.086	5
Thailand[c]	136	0.035	4
Turkey[c]	79	0.015	2
United Kingdom	90	0.017	3
Zaire[c]	136	0.026	4
Total	**4,940**	**1.033**	**162**

[a] *TRIGA is an acronym for Training, Research, Isotope, General Atomic reactors.*

[b] *To derive uranium mass in kilograms, multiply the amounts by 1,000.*

[c] *Countries with other than high-income economies, developing countries (World Bank, 1994)*

The Notice of Intent [59 Fed. Reg. 54336 (1993)] for this EIS estimated that 15,000 spent nuclear fuel elements would be accepted under the proposed action. This estimate [representing 12 MTHM, with a volume of approximately 89 m^3 (3,200 ft^3)] was prepared in early 1993, based on a projected 10-year period of generation of spent nuclear fuel at foreign research reactors in 28 foreign countries, plus the spent nuclear fuel available at these foreign research reactors as of 1993.

Since preparation of the 1993 spent nuclear fuel projection, however, cooperative understandings have been reached with several other foreign research reactor operators concerning their participation in the proposed spent nuclear fuel management program. In addition, the period of time over which the management policy would be in effect has been delayed by 3 years (to mid-1996) and thus at least 3 more years' worth of spent nuclear fuel has accumulated. Thus, the amount of spent nuclear fuel from foreign research reactors that would be accepted under the basic implementation is increased to a new total of up to 19.2 MTHM, with a volume of approximately 110 m^3 (4,100 ft^3), representing approximately 22,700 spent nuclear fuel elements of the type considered in the 1993 projection. Of this amount, approximately 4.6 MTHM is HEU, and 14.6 MTHM is LEU foreign research reactor spent nuclear fuel.

Tables 2-1 and 2-2 provide an estimate of the amount of spent nuclear fuel that has been or would be generated in each country by late 2005 (10 years from the effective date of the policy implementation), as estimated by Argonne National Laboratory based on information provided by the foreign research reactor operators (Matos, 1994). A list of the foreign research reactors included in the proposed policy is provided in Appendix B. Table 2-1 shows the inventory of aluminum-based fuel clad in aluminum, while Table 2-2 shows zirconium-based TRIGA fuel clad in either aluminum or stainless steel. These two tables are combined to yield the approximately 22,700 elements (or about 19.2 MTHM) that are estimated to be currently stored or generated by the year 2005. The tables also provide the estimated number of shipments expected from each country. The number of shipments is a key parameter in evaluating the risks associated with the handling and transportation of the foreign research reactor spent nuclear fuel.

It should be noted that the number of elements and number of shipments presented for each country in Tables 2-1 and 2-2 are estimates based on projections of the numbers of elements to be generated over a ten-year period into the future. These estimates are intended to conservatively bound the total number of foreign research reactor spent nuclear fuel elements and shipments associated with the proposed policy. However, the actual distribution of elements and shipments among the listed countries might change, within the limits of the total number of elements and shipments listed, based on actual experience gained during the lifetime of any policy that may be established.

For the purpose of analysis, the foreign research reactor spent nuclear fuel has been categorized by fuel type (aluminum-based or TRIGA) and geography (Eastern or Western) depending on the location of the likely port(s) of entry to the United States. As noted in Section 2.6.4.1, foreign research reactor spent nuclear fuel from Europe, Africa, and the Middle East and parts of Central and South America is likely to enter the United States from the east coast (Eastern) and the rest from the west coast (Western).

The distribution of the foreign research reactor spent nuclear fuel under the basic implementation is as follows:

- By Fuel Type: Aluminum-based — approximately 17,800 elements, 18.2 MTHM, 105 m^3 (3,900 ft^3)

  TRIGA — approximately 4,900 elements, 1.0 MTHM, 5 m^3 (200 ft^3)

- By Geography: Eastern — approximately 16,400 elements, 14.4 MTHM, 80 m^3 (3,000 ft^3)

  Western — approximately 6,300 elements, 4.8 MTHM, 30 m^3 (1,100 ft^3)

The assumptions used in estimating the number of shipments are included in Appendix B (Section B.1.6). Characteristics of the foreign research reactor spent nuclear fuel that would be accepted are provided in Section 2.6.1.

2.2.1.4 Location for Taking Title to Foreign Research Reactor Spent Nuclear Fuel

Under the basic implementation of Management Alternative 1, DOE would take title to the foreign research reactor spent nuclear fuel at the limit of U.S. territorial waters, or continental U.S. borders for shipments from Canada. Where DOE takes title would not have an effect on the environment. Title location of the spent nuclear fuel is relevant to questions that include the source and extent of liability for damage in the event of an accident outside the scope of Price-Anderson Act coverage. The Price-Anderson Act [42 U.S.C. §2210 (1988)] provides a mechanism by which DOE could pay for damages arising out of a nuclear incident that occurs within the United States.

2.2.1.5 Marine Transport

Under the basic implementation of Management Alternative 1, the foreign research reactor spent nuclear fuel would be shipped by chartered and/or regularly scheduled commercial ships from foreign ports to the United States. Chartered shipments would be on purpose-built ships or general purpose commercial cargo ships meeting appropriate International Marine Organization regulations. Regularly scheduled commercial shipments would be on general purpose commercial ships carrying other cargo at the same time.

Marine transport of the foreign research reactor spent nuclear fuel, as well as ground transport between ports and management sites in the United States, would be carried out in approved and certified spent nuclear fuel casks. These casks would be certified for use in the United States by the U.S. Department of Transportation if the cask was designed and fabricated in a foreign country, or by the U.S. Nuclear Regulatory Commission (NRC) if the cask was designed and fabricated in the United States. The design and fabrication of casks in a foreign country is based on the International Atomic Energy Agency standards which are the bases for those promulgated by the NRC. Marine transport activities are discussed in more detail in Section 2.6.3.

All ships entering U.S. territorial waters are required to comply with U.S. Coast Guard safety regulations and are subject to U.S. Coast Guard inspection. In addition, international transportation of hazardous material is governed by the International Movement of Dangerous Goods Code, which is one of a series of safety codes associated with the International Maritime Organization. This code establishes the international rules for shipping hazardous cargos, which includes foreign research reactor spent nuclear fuel. While most nations have agreed to follow the International Maritime Organization codes, including the International Movement of Dangerous Goods Code, compliance by individual shippers would be voluntary.

Unless DOE were to take title to the foreign research reactor spent nuclear fuel overseas (Section 2.2.2.4), the responsibility for shipping the spent nuclear fuel to the United States (if the fuel is to be accepted into the United States) belongs to the foreign research reactor operators. Under these conditions, DOE would ensure that the shipment of the spent nuclear fuel was accomplished on well-equipped, -maintained, and -operated ships through the contract that would be signed between DOE and every participating foreign research reactor operator. DOE would require the use of carriers that commit to following the International Movement of Dangerous Goods Code and all other safety requirements, such as the Safety of Life at Sea, through these contracts. If DOE were to be responsible for shipping, only shipping firms that guaranteed to follow U.S. Coast Guard regulations and international safety codes would be used to ship foreign research reactor spent nuclear fuel.

2.2.1.6 Port(s) of Entry

The basic implementation of Management Alternative 1 would involve receipt of foreign research reactor spent nuclear fuel at any of the 10 ports of entry chosen on the basis of criteria discussed in Section 2.6.3.1. All 10 candidate ports offer standard cargo container unloading services. These potential ports of entry have been identified subsequent to the application of criteria, (including appropriate experience, safe transit, adequate facilities, and population) to the universe of potential U.S. marine ports of entry. These ports are: Charleston, SC (includes Naval Weapons Station [NWS] Charleston and Wando Terminal); Galveston, TX; Hampton Roads, VA (includes Newport News, Norfolk, and Portsmouth terminals); Jacksonville, FL; Military Ocean Terminal Sunny Point (MOTSU), NC; NWS Concord, CA; Portland, OR; Savannah, GA; Tacoma, WA; and Wilmington, NC. The geographic location of each of these ports is displayed in Figure 2-3. This EIS will also assess the potential impacts of foreign

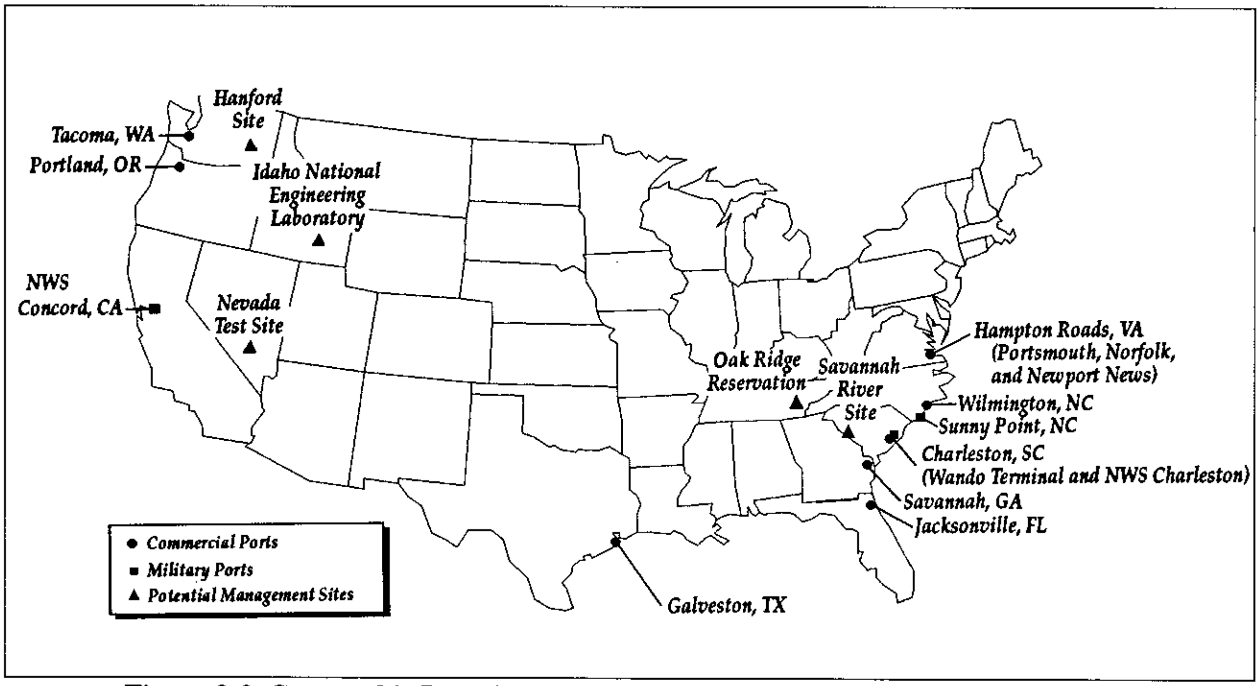


Figure 2-3 Geographic Locations of the Ports of Entry Considered for Receipt of Foreign Research Reactor Spent Nuclear Fuel

research reactor spent nuclear fuel at three high-population-density ports to bound the results of the impact analysis. These high-population-density ports are: Elizabeth, NJ; Long Beach, CA; and Philadelphia, PA. The port identification and evaluation process is discussed in Section 2.6.3 and in Appendix D.

2.2.1.7 Ground Transport

The basic implementation of Management Alternative 1 would involve shipment of foreign research reactor spent nuclear fuel from the ports of entry (both seaports and Canadian border crossings) to potential management sites. It could also involve shipment of foreign research reactor fuel between management sites. As explained in Section 2.6.4.1, the unavailability of certain sites to accept foreign research reactor spent nuclear fuel at the beginning of the management policy period would necessitate temporary receipt and management of foreign research reactor spent nuclear fuel at an available site and subsequent transportation to another site. The ground transport options and route identification process are discussed in Section 2.6.4.

Both rail and highway shipping capabilities are available at all ports of entry and each management site under consideration, with the exception of the Nevada Test Site, which has no rail capability. The shipment of foreign research reactor spent nuclear fuel was analyzed along representative highway and railway routes between all ports and the potential management sites as applicable. Barge transportation is also considered where applicable. The only management sites reasonably accessible by barge are the Savannah River Site and the Hanford Site from the ports of Savannah, GA and Portland, OR, respectively.

2.2.1.8 Foreign Research Reactor Spent Nuclear Fuel Management Sites

Potential sites considered by DOE for the receipt and management of foreign research reactor spent nuclear fuel under this EIS are the same as those considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). They are the Savannah River Site, the Idaho National Engineering Laboratory, the

Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site. Site-specific activities associated with the basic implementation of Management Alternative 1 and the implementation alternatives are discussed in Section 2.6.5. There are only two sites in the United States that could start receiving these shipments quickly: the Savannah River Site and the Idaho National Engineering Laboratory.

2.2.1.9 Storage Technologies

Under the basic implementation of Management Alternative 1, DOE would receive and manage foreign research reactor spent nuclear fuel for a period starting in approximately mid 1996, and continuing for 40 years until ultimate disposition. During the first few years, storage would take place in existing storage facilities that use both wet and dry storage technologies. For the period beyond those first few years, when construction of new facilities may become necessary, the storage technology identified for the basic implementation of Management Alternative 1 is dry storage. However, construction of new wet storage facilities is considered as an implementation alternative. Storage technologies and storage facilities considered under the basic implementation of Management Alternative 1 and implementation alternatives are discussed in Section 2.6.5.

2.2.2 Implementation Alternatives for Management Alternative 1

Environmental effects of each of the implementation alternatives are evaluated in this EIS. The range of these alternatives is presented in Figure 2-4. Chapter 4, Section 4.3, provides results of the analysis.

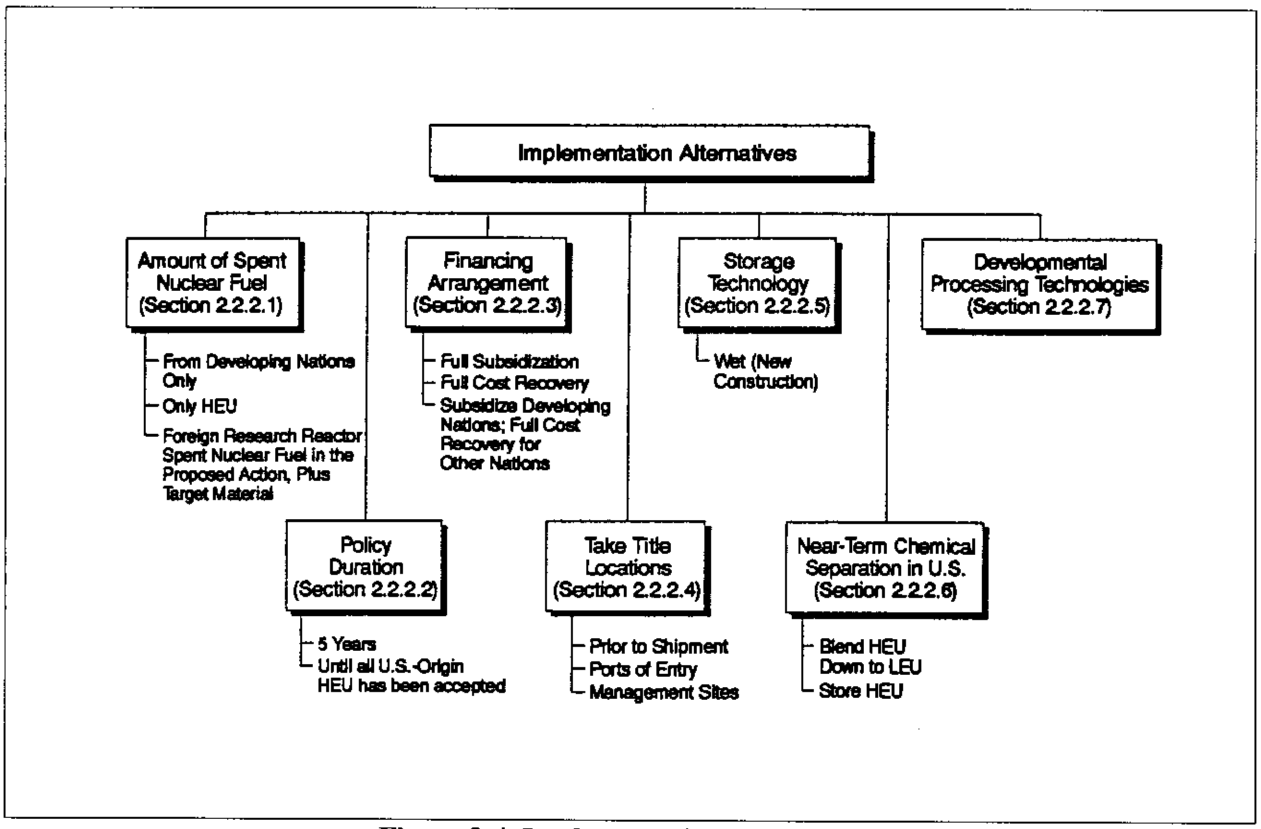


Figure 2-4 Implementation Alternatives

2.2.2.1 Implementation Alternative 1 - Alternative Amounts of Spent Nuclear Fuel to be Accepted

This implementation alternative involves choosing to accept and manage one of three subalternative amounts of foreign research reactor spent nuclear fuel:

1a. Accept spent nuclear fuel (HEU and/or LEU) only from developing countries. The foreign research reactor spent nuclear fuel from these countries contains approximately 1.9 MTHM, representing 5,000 individual elements, with a volume of 13 m^3 (500 ft^3).

1b. Accept only HEU from the research reactors eligible under the proposed action. The amount of this HEU would be approximately 4.6 MTHM, representing 11,200 elements, with a volume of 61 m^3 (2,250 ft^3).

1c. In addition to foreign research reactor spent nuclear fuel, accept HEU and LEU target materials that were used in Canada, Belgium, Argentina, and Indonesia for the production of medical isotopes. Isotope production targets[3] are irradiated in research reactors and dissolved in acid or base to extract radioisotopes that are used in medical imaging applications. The residual materials after dissolution and extraction of the radioisotopes are referred to here as target material. It is expected that this target material would contain about 0.6 MTHM, representing the uranium content of approximately 620 typical foreign research reactor spent nuclear fuel elements.

Under the last subalternative, HEU target material would be accepted until a suitable LEU target is available. After such a time, target material would be accepted from a foreign research reactor only if that foreign research reactor agrees to convert to use of LEU target.

2.2.2.2 Implementation Alternative 2 - Alternative Policy Durations

The basic implementation of Management Alternative 1 has a duration of 10 years. Two policy duration subalternatives were assessed. These are:

2a. *Five-Year Policy:*

- For foreign research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective, spent nuclear fuel (HEU and LEU) currently stored or generated during the 5-year policy period would be accepted.
- For foreign research reactors operating on HEU fuel when the policy becomes effective and agreeing to convert to LEU fuel, or having life time cores, or planning to shut down by a specific date while the policy is in effect, or for which a suitable LEU fuel is not available, HEU fuel currently stored or generated during the 5-year policy period would be accepted.
- For foreign research reactors that are already shut down, spent nuclear fuel (HEU and LEU) currently stored would be accepted.

---

3 *Canada, Argentina, and Belgium currently use aluminum-based targets containing HEU, and Indonesia currently uses a target that consists of a layer of HEU oxide material plated on the interior surface of a stainless steel tube.*

The amount of spent nuclear fuel estimated to be accepted for a 5-year policy period under this subalternative is up to approximately 18,800 individual elements containing approximately 13 MTHM, with a volume of 87 m^3 (3,300 ft^3). The distribution by fuel type and geography is as follows:

- By Fuel Type: Aluminum-based — approximately 14,100 elements; 12 MTHM, 83 m^3 (3,100 ft^3).

  TRIGA — approximately 4,700 elements, 1.0 MTHM, 4 m^3 (200 ft^3).

- By Geography: Eastern — approximately 13,400 elements, 9.5 MTHM, 65 m^3 (2,400 ft^3)

  Western — approximately 5,400 elements, 3.4 MTHM, 22 m^3 (900 ft^3)

This subalternative would allow shipments and receipt of foreign research reactor spent nuclear fuel to be made for 8 years starting from the effective date of the policy implementation, as long as the fuel had been generated within the 5-year policy period. The additional 3 years would allow for a cooldown time of fuel discharged late in the 5-year period, the logistics in arranging shipment of this fuel, as well as other possible delays from strikes, court actions, and mechanical problems.

2b. *Indefinite HEU/10-Year LEU Policy:*

- For foreign research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective, LEU spent nuclear fuel currently stored or generated in the 10-year policy period would be accepted within the time period allowed in the basic implementation (13 years). Acceptance of HEU spent nuclear fuel that had been or would be discharged from the reactor would continue indefinitely, until all such HEU spent nuclear fuel had been accepted.

- For foreign research reactors operating on HEU fuel when the policy becomes effective and agreeing to convert to LEU fuel, or planning to shut down by a specific date within 10 years of the effective date of the policy, LEU spent nuclear fuel generated in the 10-year policy period would be accepted within the time period allowed in the basic implementation (13 years). Acceptance of HEU spent nuclear fuel would continue indefinitely, until all such HEU spent nuclear fuel had been received.

- For foreign research reactors operating on HEU fuel when the policy becomes effective and for which a suitable LEU fuel is not available, or having life time cores, HEU spent nuclear fuel would be accepted:

  - from foreign research reactors with lifetime cores, until all such HEU spent nuclear fuel had been accepted, and

  - from other foreign research reactors until all HEU spent nuclear fuel at the reactor on the date the policy becomes effective, or generated within 5 years of that date, had been accepted.

- For foreign research reactors that are already shut down, the LEU spent nuclear fuel would be accepted within the period allowed in the basic implementation. The HEU spent nuclear fuel would be accepted indefinitely, until all such HEU spent nuclear fuel had been accepted.

Under this implementation subalternative, the total amount of foreign research reactor spent nuclear fuel that would be accepted is the same as in the basic implementation of Management Alternative 1.

2.2.2.3 Implementation Alternative 3 - Alternative Financing Arrangements

Under the basic implementation, the costs of participation would be fully subsidized by the United States for developing countries, however, developed countries would be charged a competitive fee.

For this implementation alternative, the cost impacts of the following subalternatives arrangements were evaluated:

- Subsidize all countries;
- Charge all countries the full cost of accepting and managing the foreign research reactor spent nuclear fuel (a full-cost recovery fee); and
- Subsidize developing countries as in the basic implementation, and charge developed countries a full-cost recovery fee.

A full-cost recovery fee would be based on the estimated cost to the United States for the safe, final disposition of the spent nuclear fuel within the United States. This fee could be based on: (1) the cost of chemically separating spent nuclear fuel and disposal of vitrified high-level waste, or (2) interim storage of the spent nuclear fuel followed by direct ultimate disposition.

In theory, this arrangement would cost the United States nothing. All costs would be borne by the foreign nations. However, many developing, and some developed nations probably would decline to pay these high costs, which could lead to HEU spent nuclear fuel stockpiled around the world, much of it remaining in the countries least able to protect it. For many countries, this arrangement would have the same impact as the No Action Alternative.

The amount of spent nuclear fuel that would be accepted under this implementation alternative would be the same as that identified under the basic implementation, or under the implementation subalternatives discussed in Section 2.2.2.1. The actual amount of spent nuclear fuel received could be less than that identified under the basic implementation because, as stated above, some countries may consider the higher fee to be a disincentive. The analysis of environmental impacts for this EIS, however, considered the amount of spent nuclear fuel to be received for this implementation alternative to be unchanged for use as an upper bounding case.

2.2.2.4 Implementation Alternative 4 - Alternative Locations for Taking Title

In the basic implementation, DOE would take title to the foreign research reactor spent nuclear fuel at the limit of U.S. territorial waters (19 km or 12 mi), or the continental U.S. border for shipments from Canada. The location for taking title is relevant to questions of liability and regulatory authority. For example, if DOE were to take title at the foreign research reactor site, there could be additional regulatory burdens on DOE, due to the laws of a particular country being imposed upon the owner of the spent nuclear fuel. The taking of title prior to shipment might impose upon DOE additional legal liability for damages not associated with a nuclear incident covered by the Price-Anderson Act. DOE and the Department of State considered the following three subalternative approaches regarding the locations for taking title to the foreign research reactor spent nuclear fuel:

- Taking title to the foreign research reactor spent nuclear fuel before shipment,
- Taking title at the port(s) of entry, and
- Taking title at the management site(s).

The amount of spent nuclear fuel that would be accepted under this implementation alternative would be the same as that identified under the basic implementation or under the implementation subalternatives discussed in Section 2.2.2.1.

2.2.2.5 Implementation Alternative 5 - Wet Storage Technology for New Construction

Under the basic implementation, storage requiring new construction would employ dry storage technology. As an implementation alternative, DOE has assessed the use of wet storage technologies for new construction, which use water-filled pools to store spent nuclear fuel. Wet storage methods have been used historically at DOE sites and by the nuclear industry.

The amount of spent nuclear fuel that would be accepted under this implementation alternative would be the same as that identified under the basic implementation or under the implementation subalternatives discussed in Section 2.2.2.1.

2.2.2.6 Implementation Alternative 6 - Near Term Conventional Chemical Separation in the United States

Under this implementation alternative, near term conventional chemical separation would be conducted at either the Savannah River Site or the Idaho National Engineering Laboratory. There are both advantages and disadvantages to chemical separation of foreign research reactor spent nuclear fuel. The advantages include the following:

- The high-level radioactive waste from the foreign research reactor spent nuclear fuel would be transformed into forms that are more suitable (i.e., more compact and stable) for storage than intact aluminum-based spent nuclear fuel.
- The high-level waste would be converted to a form that is expected to be acceptable for disposal in a geologic repository.
- Construction of some or all of the new spent nuclear fuel storage space would be avoided.
- The conventional chemical separation facilities already exist, as well as the waste treatment facilities required to put the high-level radioactive and other wastestreams in forms suitable for disposal. In contrast, there are the large technical, cost, and regulatory uncertainties associated with direct disposal of intact foreign research reactor spent nuclear fuel (much of it containing HEU).
- If disposal of intact spent nuclear fuel is shown to be technically infeasible, or if the waste acceptance criteria for a geologic repository require significant dilution of the HEU due to criticality concerns, DOE estimates that the life-cycle costs of chemical separation may be substantially lower than the cost of storage and geologic disposal of intact spent nuclear fuel. (Alternatively, if direct disposal of intact foreign research reactor spent nuclear fuel,

including that containing HEU, is shown to be technically feasible, DOE estimates that the costs of chemical separation and the storage/direct disposal option would be nearly the same.)

The disadvantages include the following:

- Chemical separation would increase the total volume of the waste (including liquid high-level waste raffinates, transuranic wastes, various solid and liquid low-level wastestreams, acidic wastes, chelating and complexing agents, and solvents). Volume reduction and other treatments would be used to prepare these wastes for disposal. (Because the requirements for direct disposal of aluminum-based spent nuclear fuel have not been established, the character and volumes of waste associated with direct disposal are uncertain.)

- The separated uranium, which DOE would prefer to blend down to LEU, would have to be stored until it could be sold or otherwise disposed of.

- The forms of the wastes generated by chemical separation are complex, involving corrosive, flammable and toxic liquids.

- The use of chemical separation by the United States as a spent nuclear fuel management technology could increase the accumulation of stockpiles of HEU unless the HEU is blended down. The United States does not engage in chemical separation for nuclear explosive purposes, and seeks to eliminate, where possible, the accumulation of stockpiles of HEU or plutonium. The United States nuclear weapons nonproliferation policy on reprocessing is summarized in the White House Fact Sheet on Nonproliferation and Export Control Policy dated September 27, 1993. A copy of this policy is included in Appendix G of this EIS.

Taking these advantages and disadvantages into account, chemical separation of foreign research reactor spent nuclear fuel in existing facilities is not preferred by DOE as a technology for routine management of spent nuclear fuel in the United States. Nonetheless, chemical separation remains a reasonable alternative in light of DOE's substantial technical experience in these operations and the availability of existing facilities.

DOE is considering the development and use of various alternatives to chemical separation for foreign research reactor spent nuclear fuel stabilization, interim storage and conditioning for disposal under Management Alternative 1, Implementation Alternative 7 in this EIS. This initiative is discussed in more detail in a DOE memorandum of December 28, 1994 from Thomas P. Grumbly to Jill E. Lytle (see Appendix G). In fact, development, demonstration and implementation of a new treatment and/or packaging technology is a key element of the preferred alternative as described in Section 2.9.

The Nuclear Waste Policy Act of 1982 (as amended) authorizes disposal of the foreign research reactor spent nuclear fuel in a geologic repository (if DOE takes title to such fuel). The foreign research reactor spent nuclear fuel, and/or the high-level radioactive waste that would result from chemical separation of the foreign research reactor spent nuclear fuel, would require storage until its disposal in such a geologic repository. A determination of whether the foreign research reactor spent nuclear fuel can be safely disposed of in a geologic repository will depend on the outcome of scientific analyses, including a repository performance analysis considering the final form in which the foreign research reactor spent nuclear fuel would be emplaced in the repository.

Near term conventional chemical separation of foreign research reactor spent nuclear fuel at the other three potential foreign research reactor spent nuclear fuel management sites was not analyzed because the Oak Ridge Reservation and the Nevada Test Site do not have facilities in which such chemical separation could be conducted, and the facilities at the Hanford Site are no longer operable. To consider chemical separation at any of these three sites, the foreign research reactor spent nuclear fuel would need to be stored in the United States during the period of time that a new chemical separation facility at one of these sites was designed, a project-specific NEPA review was conducted, and the facility constructed and put into operation. Such activities could not be completed in the near term and, accordingly, these sites were not considered reasonable alternatives for near term chemical separation.

Solid low-level radioactive waste and wastewater generated by chemical separation would be managed in the same manner as the similar wastes generated by the storage of the intact foreign research reactor spent nuclear fuel. Discussion of waste generation from storage is included in Appendix F, Section F.4. Chemical separation would also generate five types of waste that would not result from storage of intact foreign research reactor spent nuclear fuel: high-level radioactive waste, hazardous waste, mixed hazardous and radioactive waste, and low-level "saltstone" waste.

Following chemical separation of the foreign research reactor spent nuclear fuel, the resulting high-level radioactive wastes would be managed along with substantial existing inventories of identical waste. Management of the high-level radioactive wastes would include the following:

1. The high-level wastes would be transferred to storage tanks and kept there pending processing;
2. The wastes would be pretreated in preparation for further processing;
3. To preclude the necessity for transporting liquid high-level wastes, these wastes would be processed on the sites where they were generated:
    a. At the Savannah River Site, the wastes would be:
        1) Vitrified in the Defense Waste Processing Facility;
        2) The borosilicate glass resulting from vitrification would be stored pending disposal;
    b. At the Idaho National Engineering Laboratory, the wastes would be:
        1) Calcined to produce a more easily stored waste form;
        2) Stored in the calcine form pending development of a process and facility for final processing;
        3) After further research and development regarding conversion techniques and waste forms, the calcine would be converted to a form suitable for geologic disposal and stored pending disposal;
4. The final waste form would be transported to and disposed of in a geologic repository.

Transuranic wastes[4] would be stored on the site where the chemical separation would be accomplished until a permanent disposal facility, such as the Waste Isolation Pilot Plant, becomes available. Site treatment plans for low-level and transuranic mixed wastes are now being developed. Hazardous wastes would be sent to a licensed commercial treatment, storage and disposal facility. Saltstone, a mixture of low-level waste and concrete that is a by-product of high-level radioactive waste vitrification at the Savannah River Site's Defense Waste Processing Facility, would be pumped into above-ground concrete vaults onsite, where it would harden into a concrete monolith.

The Savannah River Site currently has chemical separation facilities. This capability, however, is limited to aluminum-based spent nuclear fuel. In contrast, the Idaho National Engineering Laboratory has facilities that can chemically separate both aluminum-based and TRIGA foreign research reactor spent nuclear fuel. However, these facilities would require some upgrades in order to accomplish this chemical separation. The existing dissolvers and calcination vessel could be used at the start of chemical separation activities, but would have to be replaced within a few years. A new tank farm and set of calcine bins would have to be built. Furthermore, this site does not have an existing vitrification facility or a glass waste storage building, as the Savannah River Site does. Upgrading the facilities at the Idaho National Engineering Laboratory would require additional time and funding. This EIS analyzes the impacts of chemically separating aluminum-based foreign research reactor spent nuclear fuel at both the Savannah River Site and the Idaho National Engineering Laboratory, but considers chemical separation of TRIGA foreign research reactor spent nuclear fuel only at the Idaho National Engineering Laboratory.

Under the near term chemical separation alternative, there are two components: Extent of the Chemical Separation and Uranium Disposition. Each of these components is discussed below.

Extent of the Chemical Separation

At each of the two potential sites, the foreign research reactor spent nuclear fuel could be chemically separated in a dedicated mode or as part of larger scale chemical separation activities.

Chemical Separation at the Savannah River Site Dedicated to Foreign Research Reactor Spent Nuclear Fuel: DOE would chemically separate all 18.2 MTHM of the aluminum-based foreign research reactor spent nuclear fuel, shown previously in Table 2-1. The Savannah River Site has facilities that could perform the chemical separation, so no new chemical separation facilities would need to be constructed.

DOE and the Department of State have included in this EIS analysis all 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel in Table 2-1 because this is the maximum that could be considered. It is not possible to specify how much of the aluminum-based foreign research reactor spent nuclear fuel might be chemically separated before the chemical separation facilities would have been shut down, so the analysis is based on the entire amount. If chemical separation of all the foreign research reactor spent nuclear fuel were selected, existing chemical separation facilities would be required to remain operational for the 13-year duration of receipts. Maintenance and operation of a facility dedicated solely to chemical separation of foreign research reactor spent nuclear fuel is considered to be an inefficient use of such facilities. Thus, this is a nonpreferred subalternative.

---

4 *No transuranic waste would be generated if the transuranic elements (mostly plutonium) were not extracted during chemical separation. The trace amounts of these elements that exist in the foreign research reactor spent nuclear fuel could remain in the high-level wastestream.*

Chemical Separation at the Idaho National Engineering Laboratory Dedicated to Foreign Research Reactor Spent Nuclear Fuel: DOE would restart the facilities and chemically separate all the aluminum-based and TRIGA foreign research reactor spent nuclear fuel, shown previously in Tables 2-1 and 2-2. The Idaho National Engineering Laboratory does not have all the facilities required to perform the chemical separation, so some new facilities would need to be constructed. Furthermore, DOE announced a Record of Decision on May 30, 1995 for the Programmatic SNF&INEL Final EIS. Chemical separation at Idaho National Engineering Laboratory was not included in this Record of Decision, so additional site-specific NEPA documentation would be required to restart these chemical separation facilities.

DOE and the Department of State have included in this analysis all 19.2 MTHM of foreign research reactor spent nuclear fuel in Tables 2-1 and 2-2, because this is the maximum that could be considered. It is not possible to specify how much of the foreign research reactor spent nuclear fuel might be chemically separated in this case, so the analysis is based on the entire amount. Up to approximately 12 years of operation would be required to chemically separate this amount. The construction and operation of new facilities for chemical separation dedicated to foreign research reactor spent nuclear fuel is considered inefficient, and therefore, this subalternative is not preferred.

Chemical Separation at the Savannah River Site as Part of Larger Scale Activities: DOE is in the process of preparing other NEPA reviews and making decisions that could affect the decisions to be made in this EIS. The Interim Management of Nuclear Materials Final EIS (DOE, 1995a) analyzed alternatives for stabilization of nuclear materials currently stored at the Savannah River Site that represent health and safety risks, as stored in their current forms and locations. The nuclear materials in the Interim Management of Nuclear Materials Final EIS that most closely resemble the aluminum-based foreign research reactor spent nuclear fuel are the Mark-16 and Mark-22 fuels. The preferred alternative for these fuels, as announced in the *Federal Register* (60 FR 65300), is chemical separation. Therefore, the near term chemical separation of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site as part of larger scale activities is predicated upon a decision by DOE to use the chemical separation facilities at the Savannah River Site to chemically separate the Mark-16 and Mark-22 fuels under the Interim Management of Nuclear Materials Final EIS.

The Programmatic SNF&INEL Final EIS (DOE, 1995c) considered the alternative site(s) where DOE's spent nuclear fuel (including foreign research reactor spent nuclear fuel) would be managed. DOE announced in its Record of Decision on May 30, 1995 that it intends to consolidate all its aluminum-based spent nuclear fuel at the Savannah River Site. DOE could also chemically separate other aluminum-based spent nuclear fuel that is transported to the Savannah River Site under this EIS.

The aluminum-based foreign research reactor spent nuclear fuel shown in Table 2-1 would be chemically separated along with other DOE aluminum-based spent nuclear fuel selected for chemical separation. This could amount to a maximum of 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel, 0.56 MTHM of target material under Implementation Subalternative 1c (Section 2.2.2.1), approximately 28.8 MTHM of other aluminum-based spent nuclear fuel currently stored at the Savannah River Site (Wichmann, 1995), and approximately 3.4 MTHM of aluminum-based spent nuclear fuel that could be transported to the Savannah River Site (Wichmann, 1995). In all, about 51 MTHM could be chemically separated at the Savannah River Site, requiring up to approximately 13 years of operation. The environmental impacts analysis is presented in Section 4.3.6.

Chemical Separation at the Idaho National Engineering Laboratory as Part of Larger Scale Activities: Volume 2 of the Programmatic SNF&INEL Final EIS (DOE, 1995c) includes a brief analysis of the alternative of restarting the chemical separation facilities at the Idaho National Engineering Laboratory for stabilization of nuclear materials. These facilities are currently being cleaned up in preparation for decommissioning, so restarting them would require additional site-specific NEPA documentation, but DOE is not currently performing NEPA analysis on restarting these facilities.

Volume 1 of the Programmatic SNF&INEL Final EIS (DOE, 1995c) considered the alternative site(s) where DOE's spent nuclear fuel (including foreign research reactor spent nuclear fuel) would be managed. DOE could chemically separate other spent nuclear fuel that is transported to the Idaho National Engineering Laboratory under this EIS.

All the aluminum-based spent nuclear fuel would be chemically separated along with all the TRIGA foreign research reactor spent nuclear fuel and certain other spent nuclear fuel from the Idaho National Engineering Laboratory. The amount of aluminum-based spent nuclear fuel would be approximately 51 MTHM, as described above. The additional spent nuclear fuel would include 1.0 MTHM of TRIGA foreign research reactor spent nuclear fuel and approximately 13 MTHM of TRIGA and zirconium-based spent nuclear fuel from onsite facilities (Cottam, 1995). In all, about 65 MTHM could be chemically separated under this program at Idaho National Engineering Laboratory, requiring up to approximately 12 years of operation. The environmental impact analysis is presented in Section 4.3.6.

Uranium Disposition

Chemical separation, as the name implies, would separate the uranium from the waste products. The separated LEU could be sold to the commercial sector for reuse as reactor fuel. The HEU disposition issue is being considered in a separate DOE NEPA document, the Disposition of Surplus Highly Enriched Uranium EIS. If DOE decides to blend down the HEU that is within the scope of the Disposition of Surplus Highly Enriched Uranium EIS, then the HEU that would be recovered in this implementation alternative would also be blended down. Conversely, if DOE decides not to blend down the HEU that is within the scope of the Disposition of Surplus Highly Enriched Uranium EIS, then the HEU that would be recovered in this implementation alternative would also not be blended down. Until this decision is made, however, DOE may decide to blend down HEU in specific instances. For example, DOE recently announced its decision (60 FR 65300), to blend down the HEU solutions at the Savannah River Site under the Interim Management of Nuclear Materials Final EIS.

The options for HEU disposition at the Savannah River Site are:

1. Blending it down to less than 20 percent enrichment inside the chemical separation facilities;

2. Blending it down to less than two percent enrichment inside the chemical separation facilities and then processing it to an oxide in the existing FA-Line at the Savannah River Site; and

3. Completing construction of the Uranium Solidification Facility at the Savannah River Site, then processing the HEU directly to an oxide, followed by storage in a safe, secure facility.

The options for HEU disposition at Idaho National Engineering Laboratory are:

1. Blending it down to less than 20 percent inside the chemical separation facilities, and

2. Processing the HEU directly to an oxide.

Some minor modifications to the facility would be necessary to blend down the HEU. No modifications would be necessary to process it directly to an oxide. For either option, additional NEPA documentation would be required.

Under any of the blending down options, the nuclear weapons nonproliferation goal would be satisfied because the material would not be usable in weapons after it was blended down. This material could be returned to the commercial sector and reused in nuclear reactors.

If the HEU were not blended down, DOE and the Department of State might be accused of accepting the foreign research reactor spent nuclear fuel in order to stockpile HEU for future weapons use. To address this concern, DOE and the Department of State would, over time, place the separated HEU under International Atomic Energy Agency safeguards. DOE and the Department of State have identified three possible means of implementing this International Atomic Energy Agency safeguards initiative which would require the use of a finite storage area (Material Balance Area) subject to inspections by the International Atomic Energy Agency. These are:

1. DOE could use the only available Material Balance Area: Vault 16 at the Y-12 Plant on the Oak Ridge Reservation. This vault's capacity is about 40 MTHM of HEU and it presently contains only about 10 MTHM, so the available capacity is about 30 MTHM. Chemical separation of all the spent nuclear fuel in this implementation alternative would recover less than 25 MTHM of HEU, so there is sufficient capacity in Vault 16 to store all this material.

2. A new Material Balance Area could be set up at the Savannah River Site or the Idaho National Engineering Laboratory.

3. The HEU could be stored in existing vaults, and brought out to a temporary Material Balance Area for each International Atomic Energy Agency inspection.

If a decision is made to chemically separate the foreign research reactor spent nuclear fuel, it would be DOE's preference to blend down the HEU to LEU and thus preclude the possibility of this material ever being used in a nuclear weapon.

2.2.2.7 Implementation Alternative 7 - Developmental Treatment and/or Packaging Technologies

Under this implementation alternative, DOE and the Department of State would initiate a development program that could lead to a decision to construct and operate a new facility for treatment and/or packaging of foreign research reactor spent nuclear fuel. The purpose of this potential new facility would be to treat, package and store the foreign research reactor spent nuclear fuel in a manner suitable for geologic disposal, without necessarily separating the fissile materials. Other goals of the development process would be to define a technology and facility that would operate safely, meet or exceed all applicable environmental requirements (including minimization of waste volumes, toxicity, and mobility), and be consistent with U.S. nuclear weapons nonproliferation policies.

There are numerous technologies that DOE and the Department of State could consider under such a development program. These technologies could be applied at any one of the five potential foreign research reactor spent nuclear fuel management sites, and most would require the construction of totally new facilities (although some could be implemented through modifications to existing facilities). The potential environmental impacts of the construction and operation of such a new facility cannot be estimated at this time because the technologies are still developmental and the hypothetical new facility has not been designed. Implementation of any of these technologies would require additional NEPA analysis and documentation, including additional opportunities for public review and comment.

Many of the potentially applicable technologies are already being considered under the DOE Office of Spent Fuel Management Technology Integration Working Group. A development program, such as the one that would be implemented under this alternative, is outlined in the DOE Spent Nuclear Fuel

Technology Integration Plan (DOE, 1994c). A number of these developmental technologies have progressed beyond initial technical feasibility studies and have reasonably defined cost and schedule estimates for further development. Technologies, such as the Plasma Arc Treatment Process and the Electrometallurgical Treatment Process, are being developed by the Pacific Northwest Laboratory and the Argonne National Laboratory, respectively. Other developmental technologies, however, require additional evaluation prior to undergoing detailed development efforts. Some of the development technologies and criticality prevention techniques are briefly described below. Criticality prevention is discussed in Section 2.6.1.

Developmental Treatment Technologies

Chop and Dilute: The foreign research reactor spent nuclear fuel could be rendered into shards in a mechanical chopper and added to shards of depleted uranium-aluminum alloy to prevent a criticality in the repository. The mixture would have an enrichment of no more than one percent (WSRC, 1994a).

Chop and Poison: The foreign research reactor spent nuclear fuel could be rendered into shards in a mechanical chopper and a neutron poison could be added to prevent a criticality in the repository (WSRC, 1994a). A neutron poison is an element that absorbs neutrons without fissioning, thus preventing a fission chain reaction.

Melt and Dilute: The foreign research reactor spent nuclear fuel could be melted with depleted uranium metal added to the molten mixture to prevent a criticality in the repository (WSRC, 1994a).

Melt and Poison: The foreign research reactor spent nuclear fuel could be melted and a neutron poison could be added to the molten mixture to prevent a criticality in the repository (WSRC, 1994a).

Electrometallurgical Treatment: The foreign research reactor spent nuclear fuel could be dissolved, then the aluminum could be separated from the uranium and fission products in an electrorefiner. DOE has proposed to demonstrate this process as a management option for a variety of DOE-owned spent nuclear fuel. This process would produce a mineral waste form containing most of the fission products and a metal alloy containing the rest of the fission products (DOE, 1994c).

Plasma Arc Treatment: The foreign research reactor spent nuclear fuel would be placed directly into a plasma centrifugal furnace with other material (low-enriched uranium, depleted uranium, and neutron absorbers) where it would be melted and converted into a ceramic material.

Chloride Volatility Treatment: The foreign research reactor spent nuclear fuel could be completely volatized to chlorides. This process is being investigated at the Idaho National Engineering Laboratory and would require about 15 years to develop. Then the uranium could be separated and the fission products could be converted to oxides or fluorides for vitrification (DOE, 1994c).

Glass Material Oxidation and Dissolution System: The foreign research reactor spent nuclear fuel could be converted to a glass form in this single-step process. This process was recently invented at Oak Ridge National Laboratory, and has been demonstrated at the laboratory scale. The foreign research reactor spent nuclear fuel would be melted together with glass frit and all process chemistry would occur in this molten mixture. Then, depleted uranium would be added to resolve criticality concerns before the mixture is poured into canisters (DOE, 1994c).

Dissolve and Dilute: The foreign research reactor spent nuclear fuel could be dissolved in acid and depleted uranium could be added to the solution to reduce the enrichment to no more than one percent to prevent a criticality in the repository. Then the solution would be vitrified (WSRC, 1994a).

Dissolve and Poison: The foreign research reactor spent nuclear fuel could be dissolved in acid and a neutron poison could be added to prevent a criticality in the repository. Then the solution would be vitrified (WSRC, 1994a).

Developmental Packaging Technologies

Direct Disposal in Small Packages: This is a variation of the "direct disposal"concept. The foreign research reactor spent nuclear fuel could be packed intact into small waste packages to limit the amount of fissile material in any single package. Neutron poisons would also be packed in the packages in the spaces between the fuel element plates or rods to prevent a criticality in the repository.

Can-in-Canister: The foreign research reactor spent nuclear fuel could be canned, then the cans could be encapsulated in glass inside of canisters. The encapsulation process could be performed in the Defense Waste Processing Facility at the Savannah River Site, using a high-level waste glass. In effect, the cans containing foreign research reactor spent nuclear fuel would displace an equal volume of high-level waste glass inside standard Defense Waste Processing Facility canisters.

2.3 Management Alternative 2 — Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

Under this management alternative, DOE and the Department of State would seek to encourage and facilitate the management of foreign research reactor spent nuclear fuel overseas in a manner that would be consistent with U.S. nuclear weapons nonproliferation policy. DOE and the Department of State have evaluated the following two subalternatives:

1a. Overseas Storage - Encourage and assist foreign research reactors that are able to store their spent nuclear fuel in facilities in their own countries, or in developing countries, as a step toward the final disposition of the spent nuclear fuel. U.S. assistance would be provided to ensure that appropriate storage technologies, regulations, and safeguards were applied.

In some cases, this subalternative might be implemented by expansion of the storage facilities located at the foreign research reactor sites. However, many foreign research reactor operators are associated with academic institutions with limited budgets, or have building restrictions for site-specific reasons (e.g., no physical space for expansion). Thus, the opportunities for expanded spent nuclear fuel storage at foreign research reactor sites may be limited or even nonexistent. In countries with established nuclear power programs, management might also be provided at the sites where such countries will store their power reactor spent nuclear fuel. Some foreign research reactor operators may also be able to make arrangements for indefinite storage at sites in developing countries. Ultimate disposition of the foreign research reactor spent nuclear fuel would still have to be arranged at the conclusion of the management period. In the meantime, foreign research reactor spent nuclear fuel containing HEU would be stored in up to 41 countries around the world.

1b. Overseas Reprocessing - Provide nontechnical (financial and/or logistical) assistance to foreign research reactors and reprocessors to facilitate reprocessing spent nuclear fuel overseas in facilities operated under international safeguards sufficient to satisfy U.S. nuclear weapons nonproliferation concerns. Wherever possible, the wastes resulting from this reprocessing would be returned to the country in which the spent nuclear fuel was irradiated. If the reprocessing wastes cannot be returned to the country in which the spent nuclear fuel was irradiated, such wastes might be accepted by the United States for storage and ultimate geologic disposal.

The advantages and disadvantages of the technology used for reprocessing overseas would be essentially the same as those described for chemical separation in the United States as discussed in Section 2.2.2.6.

The overseas reprocessing option will be evaluated in terms of whatever is supportive of the U.S. nuclear weapons nonproliferation policy on HEU minimization. For example, factors such as the following would have to be considered:

- An expectation that HEU separated during reprocessing would be blended down to LEU for research reactors which are converting to LEU.
- The foreign reprocessors would provide the capability to reprocess LEU as well as HEU.
- Research reactors would be encouraged to convert to LEU if an LEU fuel exists or is developed that will allow such operation.

Arrangements would have to be worked out with foreign reprocessors that would be supportive of U.S. nuclear weapons nonproliferation objectives to minimize the civil use of HEU worldwide.

Reprocessing of spent nuclear fuel is a well-established technology which is based on the same principles as chemical separation in the United States as discussed in Section 2.6.5.2, and an international commercial market has developed with a total annual capacity of several thousand MTHM (BNFL, 1994; Cogema, 1994). Large portions of this capacity are oriented toward commercial spent nuclear fuel. While these facilities are technically capable of reprocessing foreign research reactor spent nuclear fuel with relatively minor modifications [e.g., blending of the foreign research reactor spent nuclear fuel in the dissolver(s) with depleted uranium], for contractual, economic, and schedule considerations, these commercial spent nuclear fuel reprocessing facilities are less inclined to consider the potential foreign research reactor spent nuclear fuel market.

Several large (hundreds of MTHM per year) spent nuclear fuel reprocessing facilities exist for LEU spent nuclear fuel, and the annual capacity of existing facilities for research and HEU spent nuclear fuel is in the tens of MTHM range. The British facilities at Dounreay are currently capable of reprocessing foreign research reactor spent nuclear fuel. The French facilities at Marcoule are planning reprocessing of French research reactor spent nuclear fuel in the near future, and the Dounreay facility is pursuing additional contracts with foreign research reactor operators for reprocessing their spent nuclear fuel. The estimated schedule of foreign research reactor spent nuclear fuel shipments corresponds to a maximum of 2 MTHM per year, which could be accommodated by the existing reprocessing capacity for these fuels. Significantly, the most likely candidate facilities for reprocessing foreign research reactor spent nuclear fuel are located in Europe and are operated under Euratom and International Atomic Energy Agency safeguards. These facilities also offer full spent nuclear fuel management capabilities, including spent nuclear fuel storage prior to reprocessing, solidification of wastes, product and waste transportation, and assay adjustments (i.e., blending of HEU to LEU). Arrangements for shipment and disposition of the processing wastes would have to be implemented. Most processing contracts with European facilities require return of the wastes to the generator (in this case, the foreign research reactor operators) of the spent nuclear fuel.

DOE has considered the possibility of accepting, in the United States, the vitrified waste from the reprocessing of foreign research reactor spent nuclear fuel overseas. The environmental impacts from the receipt and storage of the waste in the United States are presented in Section 4.4.2.

2.4 Management Alternative 3 - Combination of Elements from Management Alternatives 1 and 2 (Hybrid Alternative)

In implementing the proposed action, DOE and the Department of State could combine implementation elements from the management alternatives analyzed in Sections 2.2 and 2.3. For example, DOE and the Department of State could consider partial storage or reprocessing overseas and partial storage or chemical separation in the United States. The impacts to the U.S. environment from hybrid alternatives would be bounded by the analysis presented in this EIS for each of the implementation alternatives for Management Alternative 1, because for each implementation alternative, the analysis considers the maximum amount of spent nuclear fuel that could be accepted, stored, and/or chemically separated in the United States.

For the purpose of illustration, DOE and the Department of State have considered an example of a hybrid alternative which is a combination of implementation elements of Management Alternatives 1 and 2. This hybrid alternative is described below.

Under this hybrid alternative, DOE and the Department of State would provide nontechnical (financial and/or logistical) assistance to foreign research reactor operators and reprocessors to facilitate the reprocessing of any foreign research reactor spent nuclear fuel at western European reprocessing facilities (i.e., Dounreay or Marcoule), as in Management Alternative 2, for foreign research reactors in countries that could accept back the reprocessed waste; and DOE and the Department of State would accept and manage the rest of foreign research reactor spent nuclear fuel in the United States, as in Management Alternative 1.

In order to comply with U.S. nuclear weapons nonproliferation policy, bilateral agreements would need to be established with the foreign governments involved to ensure that the conditions discussed in Section 2.3 for overseas reprocessing (Management Alternative 2, Implementation Alternative 1b) would be met before DOE and the Department of State would consider implementation of this hybrid alternative.

Based on the current capabilities of the reprocessors overseas, the spent nuclear fuel to be considered for reprocessing would be aluminum-based. TRIGA spent nuclear fuel could also be considered if such capability is developed; however, for the purposes of the analysis, TRIGA spent nuclear fuel would be assumed to be accepted in the United States for storage. Table 2-3 lists the countries that may be able to accept the reprocessing waste and the amount of spent nuclear fuel to be considered for reprocessing overseas. Table 2-4 shows the amount of spent nuclear fuel that would be accepted in the United States.

Under this hybrid alternative, the aluminum-based foreign research reactor spent nuclear fuel to be accepted in the United States would be chemically separated at the Savannah River Site as in Implementation Alternative 6, which is discussed in Section 2.2.2.6. The TRIGA spent nuclear fuel would be transported to the Idaho National Engineering Laboratory, where it would be stored at existing storage facilities until ultimate disposition. The distribution of the spent nuclear fuel considered in this hybrid alternative is consistent with the Programmatic SNF&INEL Final EIS (DOE, 1995c) Regionalization by Fuel Type alternative.

All the other components of this hybrid alternative are the same as the basic implementation of Management Alternative 1, specifically:

- a policy duration of 10 years with a period of acceptance of spent nuclear fuel in the United States of 13 years;

Table 2-3 Spent Nuclear Fuel Considered for Reprocessing Overseas (Hybrid Alternative Example)

Country	*Number of Elements*	*Mass (MTHM)*[a]
Belgium	1,766	0.730
France	1,962	3.442
Germany	1,504	0.909
Italy	150	0.043
Spain	40	0.016
Switzerland	159	0.128
United Kingdom	12	0.004
Total	**5,593**	**5.272**

[a] *To derive mass in kilograms, multiply by 1,000.*

Table 2-4 Amount and Distribution of Foreign Research Reactor Spent Nuclear Fuel to be Accepted in the United States (Hybrid Alternative Example)

Type	*Number of Elements*	*Mass (MTHM)*[a]	*Number of Shipments*
Aluminum-Based	12,210	12.912	406
Eastern[b]	7,593	8.647	275
Western[b]	4,617	4.263	131
TRIGA	4,940	1.033	162
Eastern[b]	3,245	0.528	107
Western[b]	1,695	0.505	55
Total	**17,150**	**13.945**	**568**

[a] *To derive mass in kilograms, multiply by 1,000.*

[b] *Refers to the location of the likely port(s) of entry to the United States.*

- a financing arrangement by which the United States would bear the full cost for transporting and managing the foreign research reactor spent nuclear fuel received from developing countries, but would charge developed countries (if any) a competitive fee;
- taking title to the foreign research reactor spent nuclear fuel at the U.S. territorial waters limit or continental U.S. borders for shipments from Canada;
- marine transport of the foreign research reactor spent nuclear fuel by chartered and/or regularly scheduled commercial ships;
- ports of entry that qualify on the bases of criteria discussed in this EIS; and
- ground transport from ports of entry to the Savannah River Site and Idaho National Engineering Laboratory by truck, rail, or barge, or a combination of these modes.

The impacts to the U.S. environment from this hybrid alternative would be bounded by the Savannah River Site portion of Implementation Alternative 6 (Near Term Chemical Separation in the United States at the Savannah River Site), which considers the acceptance of approximately 22,700 elements of foreign research reactor spent nuclear fuel in the United States versus approximately 17,100 (13.9 MTHM) elements in this hybrid alternative; the chemical separation of approximately 17,800 aluminum-based elements versus approximately 12,200 aluminum-based elements in this hybrid alternative; and the storage of approximately the same number of TRIGA elements as under the basic implementation.

The environmental impacts and policy considerations of the hybrid alternative are discussed in Section 4.5.

2.5 No Action Alternative

In the No Action Alternative, the United States would neither manage foreign research reactor spent nuclear fuel containing uranium enriched in the United States, nor provide technical assistance or financial incentives for overseas storage or reprocessing. In this case, no foreign research reactor spent nuclear fuel shipments to the United States and no assistance to foreign countries for managing foreign research reactor spent nuclear fuel overseas would take place. The No Action Alternative would have environmental impacts outside the United States which are not in the scope of this EIS. Policy considerations are discussed in Section 4.6.

2.6 Characteristics of the Components of the Basic Implementation

This section summarizes information on the selection process for some of the components of the basic implementation, as well as characteristics, assumptions and physical parameters used in the environmental impact analysis. This section provides characteristics of the spent nuclear fuel to be received, the types of transportation casks considered, the marine ports considered and method of their identification, the ground transportation routes considered, method of identification, typical characteristics of the wet and dry storage technologies, descriptions of the designs of facilities for both dry and wet storage technologies, descriptions of chemical separation facilities, and details on the site-specific options in managing the foreign research reactor spent nuclear fuel.

2.6.1 Characteristics and Types of Foreign Research Reactor Spent Nuclear Fuel

Spent nuclear fuel is fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated. Fuel in a reactor consists of fuel elements that come in many configurations; but generally consist of the fuel matrix, cladding and structural parts. The matrix, which contains the fissionable material is typically in the form of plates or cylindrical pellets. The cladding (typically zirconium, aluminum, or stainless steel) surrounds the fuel, confining and protecting it. Structural parts (generally nickel alloys, stainless steel, zirconium, or aluminum) hold the fuel elements in the proper configuration in the reactor core.

Spent nuclear fuel is radioactive because of the presence of the radioactive isotopes, which are products of the fission process. The radiation of most concern from spent nuclear fuel is gamma rays. Although the radiation levels can be very high, the gamma ray intensities are readily reduced by shielding the spent nuclear fuel elements with such materials as steel, lead, concrete, and water during the various management phases of handling, transporting, or storing the spent nuclear fuel elements.

An issue associated with the management of spent nuclear fuel containing significant amounts of fissionable material is the potential for a self-sustaining nuclear fission process called criticality. Prevention of criticality conditions enters in the design of the spent nuclear fuel transportation casks, the spent nuclear fuel storage and processing facilities, and the spent nuclear fuel packaging for ultimate disposition. In general, criticality prevention is accomplished by either controlling the amount of fissionable material present within a certain volume (dilution or spatial separation techniques) or by introducing neutron absorbing materials that reduce the number of neutrons available to the fission process (poisoning technique). The criticality issue has been addressed in all implementation alternatives considered for the management of the foreign research reactor spent nuclear fuel.

Two types of foreign research reactor spent nuclear fuel are covered under the proposed action. They are aluminum-based fuel and TRIGA reactor type fuel. The aluminum-based fuel refers to fuels that consist of an alloy of uranium and aluminum, or a dispersion of uranium-bearing compound in aluminum, both clad in aluminum. The enrichment of uranium can be either HEU or LEU. Details on the physical and nuclear characteristics of the aluminum-based foreign research reactor spent nuclear fuel can be found in Appendix B. The aluminum-based fuels are used in various reactor types in different forms and geometries. The spent nuclear fuel element geometries are either cylindrical, boxed type, annular with hundreds of involute plates, or pin cluster. The aluminum-based fuel forms are either plates, tubes, rods, or pins. The ^{235}U[5] content of a fuel element prior to irradiation in a reactor (i.e., fresh fuel element) can vary from about 3 g (.11 oz) to about 8,500 g (300 oz). The length of an individual element can vary from 22 cm (8.7 in) to about 300 cm (118 in).

The TRIGA reactor fuel uses uranium-zirconium hydride (U-Zr-H_x) fuel material in which the hydrogen moderator is homogeneously contained within the fuel material. The initial ^{235}U content of each rod varies between 38 g (1.3 oz) and 133 g (4.7 oz). The overall length of a TRIGA fuel rod is approximately 76 cm (30 in), and the weight is between approximately 1 kg (2.2 lb) and about 4 kg (8.8 lb). Details on the physical and nuclear characteristics of the TRIGA foreign research reactor spent nuclear fuel can be found in Appendix B.

In contrast, a typical nuclear power reactor fuel (e.g., pressurized water reactor fuel) is three to five percent enriched uranium-oxide. The fuel form is ceramic pellets combined into rods, and the cladding is zircaloy or stainless steel. A typical pressurized water reactor fuel assembly weighs 682 kg (1,500 lb) and has a length of 389 cm (13 ft). Figure 2-5 graphically depicts the differences in size of a typical pressurized water reactor assembly, a typical aluminum-based fuel element, and a TRIGA fuel element. Additional detailed information on the aluminum-based and TRIGA fuels are provided in Appendix B of this EIS.

In addition to aluminum-based and TRIGA-type spent nuclear fuel, target material containing HEU is considered for management under Implementation Alternative 1, subalternative 1c (Section 2.2.2.1). Targets are irradiated in a research reactor to produce molybdenum-99, a medical isotope. Molybdenum production peaks at a low burnup, about three percent. Once the target is removed from the reactor, the fuel is dissolved in acid, and molybdenum-99 is separated from the solution. The residual material after removal of molybdenum-99 is called target material, and is currently kept in solution form. The target material considered for management would be put in U_3O_8 or UO_2 form and canned for transport to the United States. It is expected that the target material would contain about 0.6 MTHM (the uranium content of 620 typical Material Test Reactor [MTR] elements) and a volume of 6.5 m^3 (229.5 ft^3). This material could be brought to the United States in cans having a cavity of 6.4 cm (2.5 in) in diameter and 28 cm (11 in) long, and containing between 40 g to 200 g (1.41 oz to 7 oz) of ^{235}U each.

2.6.2 Transportation Casks

Spent nuclear fuel elements are transported in stainless steel packages called transportation casks, or just casks. A typical cask for the transportation of foreign research reactor spent nuclear fuel elements is shown in Figure 2-6. Detailed descriptions of typical casks are provided in Appendix B (Section B.2).

---

5 *During the enrichment process, the amount of fissionable Uranium-235 (^{235}U) is increased. Uranium increased to less than 20 percent ^{235}U is called LEU. Uranium enriched to 20 percent or greater ^{235}U is called HEU.*

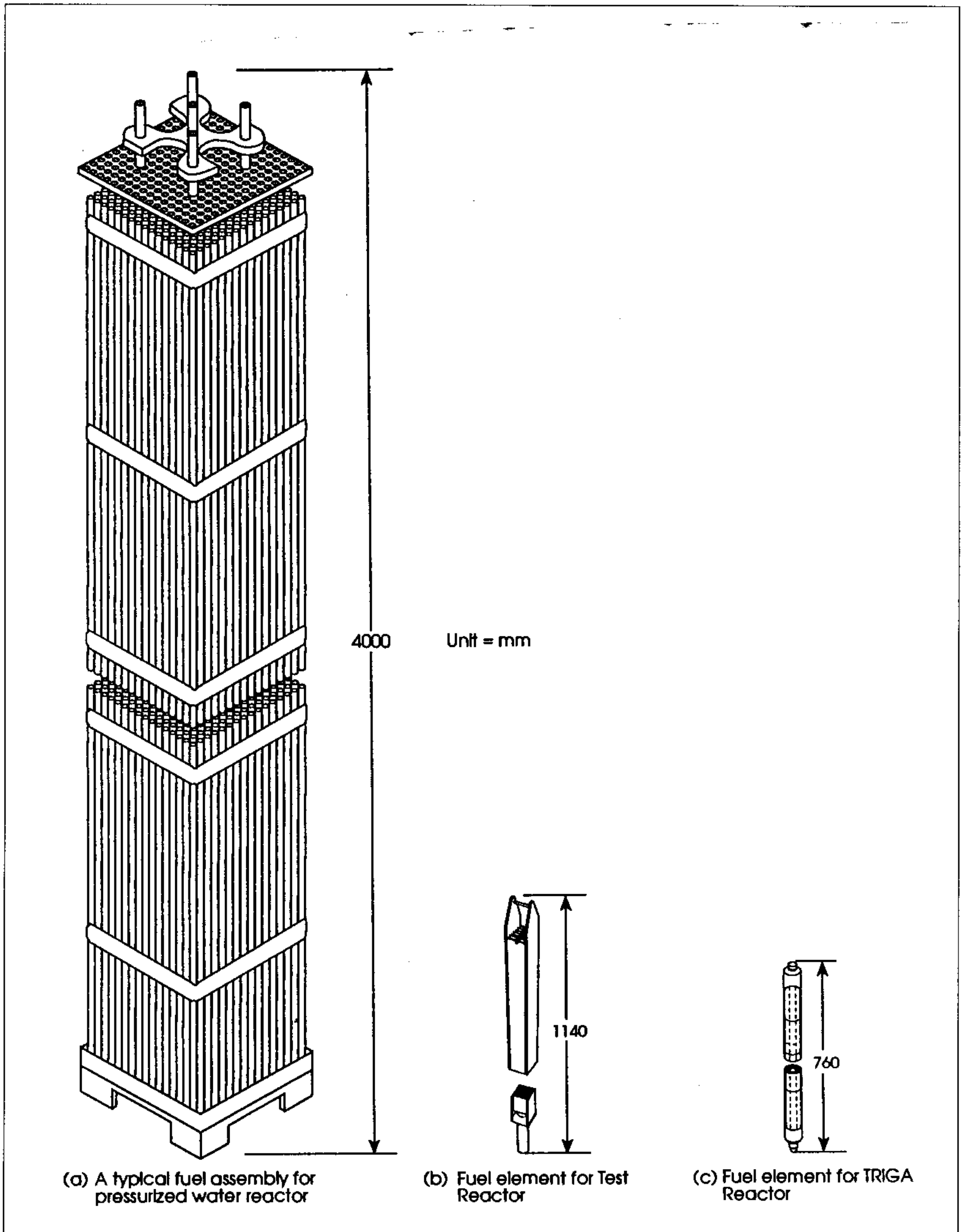


Figure 2-5 Typical Spent Nuclear Fuel Elements

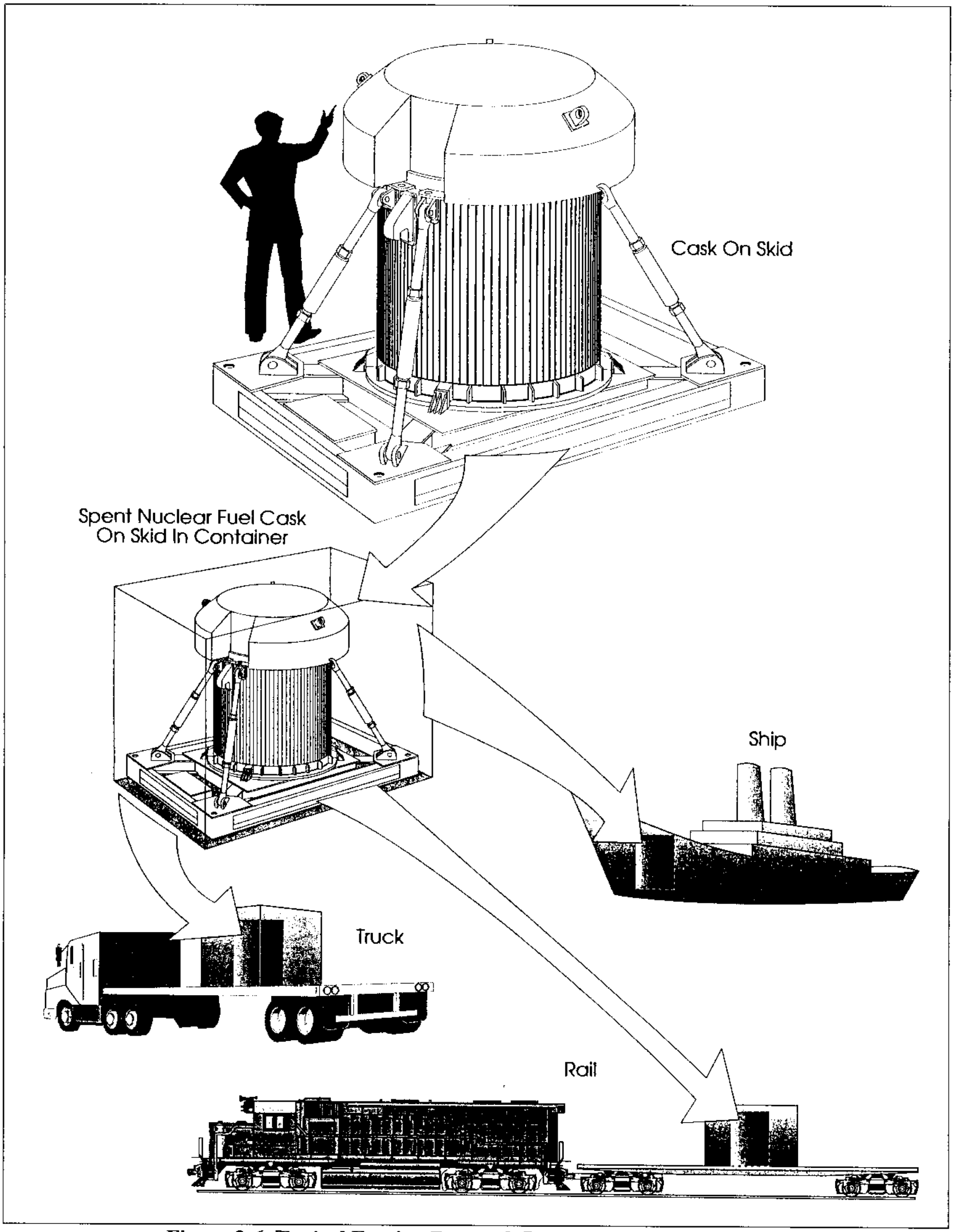


Figure 2-6 Typical Foreign Research Reactor Spent Nuclear Fuel Transportation Cask

Table 2-5 Representative Transportation Casks for Foreign Research Reactor Spent Nuclear Fuel

Transportation Cask	*Country of Origin*	*Certificate of Compliance*	*Estimated Capacity Number of Elements*
Marine Transport			
LHRL-120	Australia	USA/0389/B(U)F	114
GNS-11	Germany	USA/0381/B(U)F	21-33
TN-1	Germany	USA/0316/B(U)F	126
IU-04	France	USA/0100/B(U)F	36-40
TN-7 (TN-7/2)	Germany	USA/0130/B(U)F	60-64
NAC-LWT	United States	USA/9225/B(U)F	48-64
UNIFETCH	United Kingdom	GB/1113/B(M)F	24-40
GOSLAR	Germany	USA/0094/B(M)F	13
Ground Transport (Between Sites)			
NL1-10/24	United States	USA/9023/B( )F	120-160
IF-300	United States	USA/9001/B( )F	84-112
BMI-1	United States	USA/5957/B(U)F	24
GE-2000	United States	USA/9228/B(U)F	24
TN-8	Germany	USA/9015/B()F	36-48
NLI-1/2	United States	USA/9010/B( )F	48-64
NAC-LWT	United States	USA/9225/B(U)F	48-64

A full cask can carry from 13 to 120 spent nuclear fuel elements from foreign research reactors, depending on fuel element design, size, and cask capacity. The casks are certified as "Type B" under regulations. To receive this certification, a cask must successfully pass tests simulating severe accident conditions. The tests include being dropped onto an unyielding surface, dropped onto a steel post, subjected to extremely high temperatures of 802°C (1,475°F) for 30 minutes, and submersion in water for 8 hours.

As discussed in Section 2.7.2, "Type B" casks have been used for years to transport spent nuclear fuel elements within the United States and around the world (DOE, 1994d). To date, no spent nuclear fuel transportation cask has ever been punctured or released any of its radioactive contents, even in actual highway accidents (NRC, 1993).

The casks are designed to provide shielding from radiation. However, a low radiation field is present outside the cask — usually less than one mrem per hour at 1 m (3.3 ft) away from the cask.

Table 2-5 identifies typical transportation casks that could potentially be used for transporting foreign research reactor spent nuclear fuel from the foreign research reactor sites to the candidate United States ports of entry and to the potential management sites. A majority of these have already been used by DOE for transporting foreign research reactor spent nuclear fuel.

As explained in Section 2.6.4, the inability of certain management sites to accept foreign research reactor spent nuclear fuel at the beginning of the implementation period could necessitate temporary (as long as 10 years from the start of the policy) management of foreign research reactor spent nuclear fuel at an available site and the eventual transport of this foreign research reactor spent nuclear fuel to another site. At the time of the intersite transport, the foreign research reactor spent nuclear fuel elements would contain less radioactivity and less heat because of the decay process. The transportation, therefore, could be carried out in casks with larger capacity than those considered for marine transport. Such casks, currently licensed only for the transportation of commercial spent nuclear fuel in the United States, would need to be certified for foreign research reactor spent nuclear fuel. The size of these casks would allow the transport of up to 4 (truck-size casks) or up to 10 (rail-size casks) times as many elements in a single shipment as those considered for the marine transport casks. The bottom part of Table 2-5 identifies typical casks for intersite transportation. Description and design information is included in Appendix B (Section B.2).

2.6.3 Marine Transport and Ports

This section describes the potential activities related to foreign research reactor spent nuclear fuel marine port identification and marine transport activities.

2.6.3.1 Marine Port Identification

In this EIS, port screening and selection were performed to identify candidate ports of entry for the foreign research reactor spent nuclear fuel. The criteria used in this process were based on several sources, including:

- A DOE-sponsored workshop on port selection criteria for spent nuclear fuel held at the U.S. Merchant Marine Academy at Kings Point, NY, on November 15-16, 1993 (USMMA, 1994).
- Public input to the scoping meetings for this EIS, as summarized in the DOE Implementation Plan (DOE, 1994h).
- Factors identified in Section 3151 of the National Defense Authorization Act for Fiscal Year 1994.

These sources are described in more detail in Appendix D. After consulting the above-mentioned sources, a list of criteria for ports eligible to receive spent nuclear fuel was developed. These criteria are:

- Appropriate port experience - port terminal(s) and operators should routinely handle containerized dry cargoes that require the same type of handling as containerized spent nuclear fuel, or will have the capability to handle these cargo types during the proposed management policy period;
- Port transit - the port should be within reasonable distance from the open sea, with a good ship channel;
- Appropriate port facilities - the port should have adequate crane(s), piers, and depth of water alongside the pier;
- Ready intermodal access - the port should have ready access for intermodal transport; and
- Low human population - the human population of the ports and along transportation routes to potential management sites should be low to the extent feasible and maximum extent practicable.

These criteria, taken collectively, provided DOE and the Department of State with the basis for identifying and analyzing potential ports of entry. Additionally, port identification was expanded (i.e., the NWS Charleston was added to the Port of Charleston) as a result of public comment on the Draft EIS.

2.6.3.2 Marine Transport and Port Activities

2.6.3.2.1 Marine Transport

DOE and the Department of State estimate that approximately 721 cask loads of foreign research reactor spent nuclear fuel would be sent to the United States by ship over the 13-year acceptance period under the basic implementation of Management Alternative 1. The International Maritime Organization currently limits the typical commercial cargo ship (Class INF-2) to a maximum of 200 petrabecquerels of radioactivity (IMO, 1993), which equates to approximately 5.4 million Ci. A typical cask of foreign research reactor spent nuclear fuel is predicted to contain 1 million Ci (see Appendix C). Therefore, a shipment in a commercial cargo ship could contain several casks.

Two types of analysis were conducted to evaluate the impacts of the marine transport of foreign research reactor spent nuclear fuel: first, assuming there are no accidents (incident-free); second, assuming various accidents occur. The incident-free analyses were conducted for ships' crews and port workers, assuming ships carrying two and eight casks of foreign research reactor spent nuclear fuel. Accident analyses were conducted for accidents in port and for accidents in coastal waters and the open ocean. The number of shipments is a parameter of primary importance in the incident-free analysis as well as the accident analysis in port, coastal waters, and open ocean. As noted above, 721 shipments were considered for the basic implementation of Management Alternative 1. In implementing the proposed policy, DOE would attempt to minimize the number of shipments by maximizing the number of casks that would be carried in a single shipment. However, for the purpose of assessing the environmental impacts, a single-cask per shipment assumption is made for the purpose of conservatism. The number of shipments for the implementation alternatives discussed in Sections 2.2.2.1 through 2.2.2.7 and Management Alternative 3 discussed in Section 2.4 are roughly proportional to the amount of foreign research reactor spent nuclear fuel to be accepted in the United States under each alternative. The exact number of shipments assumed in the analysis is provided in Appendices C and D, Section C.4 and D.4, respectively. The results of both the incident-free and the accident analyses are presented in Chapter 4, with details in Appendices C and D.

There are four types of ships that could be used to transport foreign research reactor spent nuclear fuel casks. These are:

Container vessels: These are typically large ships specifically intended for the transport of containerized cargo. Some modern container ships can transport up to about 5,000 containers, although a more typical capacity is in the range of 800 to 1,000 containers. A principal advantage of container ships, because of standardization of containers, is that the vessel can be rapidly loaded or off-loaded at those ports equipped with container gantry cranes. Containers can be removed from, or placed on, the vessel at an average rate of about 45 containers per hour. At well-equipped container ports, two cranes are used to move containers.

Roll-on/roll-off ships: These ships are vehicle carriers used for the ocean transport of cars and trucks. The ships are loaded and unloaded using a ramp between the vessel and dock. Typically, the vessel carries its own ramp, which is deployed by an on-board crane, hydraulic cylinders, or chain drives. The ramp may extend from the stern of the vessel or from a hatch in the side of the vessel hull. At docks intended for roll-on/roll-off service, additional ramps may be deployed from the dock to expedite loading or unloading. This type of ship could carry foreign research reactor spent nuclear fuel casks secured on trailers.

General/cargo (breakbulk) ships: General cargo vessels are smaller ships that typically call on less well-developed or equipped ports. They have on-board jib or boom type cranes that can be used to load or unload the ship if dockside crane service is not available. As the name implies, these vessels are intended to accommodate a wide variety of cargoes that may have any reasonable configuration. Since the advent of the widespread use of containers, most of these ships are equipped with lock fixtures to secure containers during transport. If necessary, containers can be lifted on and off these ships by using four-legged slings between the corners of the container and hook of the crane.

Purpose-built ships: For the purposes of this EIS, the ships discussed here are specifically designed to transport spent nuclear fuel casks. These ships are not used for the transport of any other cargo, and they operate as chartered vessels. Casks are loaded directly into the holds of the ship because the cargo compartments contain the hardware needed to mate with the tiedown fixtures of the cask. If the ship has no crane, dockside cranes are used for loading and unloading. The cargo compartments are typically intended to handle only one cask type, however, other casks may be used with minor modifications. For the relatively efficient transport of spent nuclear fuel, the casks are large. These type vessels are intended for the transport of commercial power nuclear reactor fuel, and they generally operate between nuclear installations (power plants and spent nuclear fuel end-use facilities) having dedicated docks. Commercial docks are not normally used, but could be. These vessels have double bottoms and hulls and collision damage-resisting structures within the hull. The vessel crew is trained in the handling of the cargo and in emergency response like most other commercial vessels.

The potential exists that spent nuclear fuel would be accepted from all 41 countries that have expressed interest in this program. Ships carrying the foreign research reactor spent nuclear fuel would follow normal shipping routes from a convenient port in or near the country of origin, and would go to a U.S. port that is consistent with the port identification, evaluation, and selection process as described in Appendix D.

Regularly scheduled commercial service cargo ships could be used to ship foreign research reactor spent nuclear fuel. Some, if not most, of the regularly scheduled commercial ships might initially call at a port other than the port of destination of the foreign research reactor spent nuclear fuel, and may make additional stops. Therefore, marine transport may involve entry into and departure from intermediate ports and shipping in coastal waters. Typically, ships spend 1 day in each port of call and 1 or 2 days passing between ports.

Risks to the ships carrying foreign research reactor spent nuclear fuel and to the spent nuclear fuel itself can arise from natural sources, such as storms at sea, and from other events, such as collisions with other ships and marine obstacles, as well as from fires. Modern technology and good communications help minimize these risks by keeping ships informed of severe weather and other shipping and marine obstacles. Risk to the cargo is further reduced through proper stowing and securing, and through daily cargo inspections while at sea to ensure that the cargo remains secured.

Regardless of the technology and practices mentioned above, accidents involving ships carrying foreign research reactor spent nuclear fuel would be possible. Consequences of accidents at sea have been evaluated and are discussed in Chapter 4, and described in more detail in Appendix C.

The presence of a cask containing foreign research reactor spent nuclear fuel onboard could result in a radiation dose to some of the ship's crew due to radiation that emanates from the cask. Most of the ship's crew would be relatively far away from the cargo (and the cask), and therefore, would receive essentially no radiation dose. However, the daily inspection of the cargo would bring an inspector in close proximity to the cask containing the foreign research reactor spent nuclear fuel for a short period of time. Effects on inspectors and other incident-free impacts were evaluated, and are described in Chapter 4 and detailed in Appendix C.

Both commercial and military ports were evaluated for potential use as ports of entry for the foreign research reactor spent nuclear fuel. DOE determined that the security provisions specified by 10 CFR 73, which are required for all spent fuel shipments, could be implemented at either commercial or military ports. Any additional security that might be available at a military port would not be required for foreign research reactor spent nuclear fuel shipments.

2.6.3.2.2 Port Activities

Entry into a port by commercial vessels is accomplished under the control of the port authority. The port authority is responsible for the area from the sea buoy to the dock, except where the approaches are long, such as the case with the Chesapeake Bay. Normally, each ship is required to have a pilot familiar with local conditions to direct it while underway in the harbor or channel and during both entry and exit approaches. The pilot's job is to ensure that the ship follows the marked channel and arrives safely at its dock or other assigned location. In the event of bad weather or low visibility, radar and other instruments are available on all ships that would be considered for carrying foreign research reactor spent nuclear fuel.

In most cases, the ship moves directly to its assigned dock. However, if the assigned dock is still occupied or is not immediately available for other reasons, the ship may anchor in the harbor or its approaches for a period of time prior to docking. Most ocean-going vessels are not highly maneuverable in confined spaces, so docking is normally accomplished with the help of one or more tugs.

At the ship's first port of call in the United States, the U.S. Coast Guard and other authorities would inspect the ship, its cargo, and documentation. In the case of radioactive cargoes, the NRC may inspect the container with the radioactive material and its documentation. State and local officials could also perform inspections of documents and cargo. At ports of call after the initial port(s) of entry, additional inspections may be performed by Federal, State, or local officials.

Except for roll-on/roll-off ships, all cargo ships that would potentially carry foreign research reactor spent nuclear fuel are unloaded with cranes. Unloading of a foreign research reactor spent nuclear fuel cask, whether in a container or not, involves connecting a lifting fixture to the container or to the cask pallet, lifting the container or cask, and placing it dockside, either on an intermediate vehicle or directly on the primary mode of land transportation. Typically, container ships can be unloaded at the same rate as they are loaded (approximately 45 containers per hr), while unloading a cask on a handling platform in a breakbulk ship would require more time.

All shipments of foreign research reactor spent nuclear fuel would be anticipated well in advance, so the container housing the foreign research reactor spent nuclear fuel cask would normally be loaded immediately on the ground transportation to be used to carry it out of the port. Should there be a delay, the container may be temporarily stored at the port for up to 24 hours. Port security at any of the ports selected for analysis is adequate to protect the container in the event of this unexpected delay.

In spite of all of the precautions taken, accidents in the port would be possible. In fact, most ship accidents occur in or around ports. DOE has had no radioactivity released in the past due to port accidents; however, a range of accidents, both at the dock and in the port or its approaches, has been evaluated. See Chapter 4 for a discussion of the results of these analyses.

All ports considered for receiving foreign research reactor spent nuclear fuel would have emergency plans for responding to an accident in the port.

2.6.4 Ground Transport Route Options and Route Identification Process

2.6.4.1 Ground Transport Route Options

Route options for the potential ground transportation of foreign research reactor spent nuclear fuel depend on the marine ports considered, the management sites and the various ways that foreign research reactor spent nuclear fuel would be distributed among the potential management sites according to the alternatives considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). Accordingly, routes and the amount of fuel to be shipped would be established based on one of the following spent nuclear fuel distributions:

- an even distribution of foreign research reactor spent nuclear fuel between the Savannah River Site and the Idaho National Engineering Laboratory under the Decentralization and the 1992/1993 Planning Basis alternatives;
- a distribution that sends TRIGA spent nuclear fuel to the Idaho National Engineering Laboratory and aluminum-based spent nuclear fuel to the Savannah River Site under the Regionalization by Fuel Type alternative;
- a distribution that sends the spent nuclear fuel entering the United States from the Eastern ports to the Savannah River Site or the Oak Ridge Reservation and the spent nuclear fuel entering the United States from the Western ports to the Idaho National Engineering Laboratory, the Nevada Test Site, or the Hanford Site under the Regionalization by Geography alternative; or
- a distribution that sends all foreign research reactor spent nuclear fuel to one of the five potential management sites under the Centralization alternative.

For the purposes of this EIS, the distribution of foreign research reactor spent nuclear fuel between sites under Regionalization by Geography and by Fuel Type has been analyzed in detail. The more detailed planning performed in preparation for the analyses of the various alternatives considered in this EIS did not reveal any physical situation in which an even distribution of the spent nuclear fuel between two sites was advantageous. Furthermore, the impacts of activities associated with the even distribution at either site would be bounded by and equal to roughly 50 percent of the impacts of the centralization of all foreign research reactor spent nuclear fuel management activities at that site. As a result, this alternative is not analyzed in detail in this EIS.

An additional factor which would affect the route options for ground transportation is the inability of certain potential spent nuclear fuel management sites to implement the foreign research reactor spent nuclear fuel management policy immediately. Of the five sites, only two (the Savannah River Site and the Idaho National Engineering Laboratory) would be immediately available in late 1995. The other three could become available at a later date when appropriate facilities for accepting and managing foreign research reactor spent nuclear fuel become available. This constraint affects the ground transportation route options in the case that a site, other than the Savannah River Site or the Idaho National Engineering Laboratory, is considered and for DOE's spent nuclear fuel management under either the Regionalization by Geography or the Centralization alternative. If the Nevada Test Site, the Oak Ridge Reservation, or the Hanford Site is one of the management sites, the foreign research reactor spent nuclear fuel would have to be shipped first to one of the available management sites (the Savannah River Site and/or the Idaho National Engineering Laboratory) and later, when appropriate facilities are completed, to either the Nevada Test Site, the Oak Ridge Reservation, or the Hanford Site.

Certain assumptions are required in order to simply and consistently describe the manner in which foreign research reactor spent nuclear fuel would be transported to the management sites. The shipments, which were identified earlier in Tables 2-1 and 2-2, were divided into east coast and west coast shipments, depending on the country of origin. Spent nuclear fuel shipments from Europe, Africa, the Middle East, and parts of Central and South America were designated as east coast shipments. All others were designated as west coast shipments. Shipments from Canada were assumed to enter the United States from either an eastern or western point of entry, depending on the point of origin in Canada. Under these assumptions, for the basic implementation of Management Alternative 1, the Eastern points of entry would receive 651 cask shipments (535 from ports, 116 from Canada) and the Western ports of entry would receive 186 cask shipments (all from ports).

No intersite shipments would be necessary under the Programmatic SNF&INEL Final EIS alternatives (DOE, 1995c) that use the Savannah River Site and/or the Idaho National Engineering Laboratory for managing the foreign research reactor spent nuclear fuel. The estimated number of shipments for the basic implementation of Management Alternative 1 in these cases would be as follows:

- Decentralization, 1992/1993 Planning Basis, or Regionalization by Geography to the Savannah River Site and the Idaho National Engineering Laboratory - the Savannah River Site would receive 651 casks from the east coast and the Idaho National Engineering Laboratory would receive 186 casks from the west coast.

- Regionalization by Fuel Type - the Savannah River Site would receive 675 casks of aluminum-based fuel; 544 from the east coast and 131 from the west coast. The Idaho National Engineering Laboratory would receive 162 casks of TRIGA-type fuel; 107 from the east and 55 from the west.

- Centralization to the Idaho National Engineering Laboratory or Centralization to the Savannah River Site - the site would receive 837 casks; 651 from the east coast and 186 from the west coast.

A two-phased program would be required if a site other than the Idaho National Engineering Laboratory or the Savannah River Site is considered as a central or regional site. Phase 1 is defined as the period from the beginning of the policy (late 1995) until the Phase 2 site (the Hanford Site, the Nevada Test Site and/or the Oak Ridge Reservation) would be ready to receive fuel, which is estimated to be 10 years for new construction; less time would be required for refurbishment of an existing facility. During Phase 1, DOE would manage the fuel at the Savannah River Site and/or the Idaho National Engineering Laboratory. During Phase 2, DOE would ship any fuel that is being managed during Phase 1 at a non-Phase 2 site to a Phase 2 site, and manage the fuel at that site until a repository becomes available. The phases are defined to help describe the implementation of the foreign research reactor spent nuclear fuel management policy and to analyze the transportation impacts of the implementation of the policy.

If the Hanford Site, the Nevada Test Site, and/or the Oak Ridge Reservation were selected under the Programmatic SNF&INEL EIS, DOE and the Department of State would select from the following four strategies for managing fuel at the Savannah River Site and/or the Idaho National Engineering Laboratory during Phase 1. DOE could: (1) divide the fuel by geography, (2) divide the fuel by type (aluminum-based and TRIGA), (3) ship all fuel to the Savannah River Site, or (4) ship all fuel to the Idaho National

Engineering Laboratory. Therefore, in Phase 2, the Hanford Site and the Nevada Test Site could receive all foreign research reactor spent nuclear fuel, or TRIGA or Western spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1. Similarly, the Oak Ridge Reservation could eventually receive all foreign research reactor spent nuclear fuel or the aluminum-based or Eastern spent nuclear fuel managed at the Savannah River Site during Phase 1.

An assumption on the rate at which spent nuclear fuel arrives is necessary to estimate the number of shipments that would arrive during Phases 1 and 2. The demand to ship fuel by the foreign research reactor operators would be highest at the beginning and the end of the proposed policy period. However, the limited availability of casks would compel DOE to receive fuel at a steady rate. DOE could control the rate at which fuel is delivered by managing the contracts with shippers. Therefore, for the purposes of this analysis, it would be reasonable to assume that the 837 casks would arrive at a uniform rate during a 13-year period. Based on this rate of about 65 casks per year, it is estimated that 644 casks would be received during Phase 1 (approximately 10 years), and 193 casks would be received during Phase 2.

The projected number of shipments for the two-phased regionalization and centralization approaches are shown in Tables 2-6 and 2-7. The projections are based on the types and locations of spent nuclear fuel described in Appendix B, and the strategies and arrival rate assumptions described above. Each projection is described in more detail and shown on a map in Appendix E.

As noted earlier, the impact analysis from transportation depends on the location of entry (Eastern or Western ports) and number of shipments that would reach the United States. The discussion above pertains to the basic implementation of Management Alternative 1. In considering the implementation alternatives discussed in Sections 2.2.2.1 through 2.2.2.7 and Management Alternative 3, discussed in Section 2.4, both the number of shipments and locations of entry would vary with each alternative. The detailed distribution and number of shipments assumed to set up the ground transportation routes for each alternative are provided in Appendix E, Section E.8.

2.6.4.2 Route Analysis

Foreign research reactor spent nuclear fuel shipments would have to comply with both NRC and Department of Transportation regulatory requirements. The highway routing of spent nuclear fuel is systematically determined in accordance with Department of Transportation regulations [49 CFR 171-179 and 49 CFR 397].

The Department of Transportation routing regulations require that these shipments be transported over a preferred highway network including:

- Interstate highways;
- An interstate system bypass or beltway around a city; or
- State-designated preferred routes.

The selection of the preferred highway routes are consistent with the U.S. Department of Transportation's published guidelines (DOT, 1992).

In addition to defining routes, 49 CFR Part 397 contains the driver safety requirements for highway carriers of packages of radioactive material exceeding a quantity of material known as a "highway route-controlled quantity." All spent nuclear fuel shipments would be expected to exceed this quantity.

Table 2-6 Shipment Summary for Regionalization by Geography Alternatives

Spent Nuclear Fuel Site Option Western/Eastern	*Phase 1 Approach*	*Phase 1 Port-to-Site Shipments*	*Site-to-Site Shipments*[a]	*Phase 2 or Port-to-Final Site Shipments*	*Total Number of Shipments*[a]
INEL/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51	East to ORR: 150 West to INEL: 43	963/888
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52	East to ORR: 150 West to INEL: 43	967/889
	All to INEL	644	None	East to ORR: 150 West to INEL: 43	837
NTS/SRS	Geographic	East to SRS: 501 West to INEL: 143	INEL to NTS: 36/15	East to SRS: 150 West to NTS: 43	873/852
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	INEL to NTS: 31/13	East to SRS: 150 West to NTS: 43	868/850
	All to SRS	644	None	East to SRS: 150 West to NTS: 43	837
NTS/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51 INEL to NTS: 36/15	East to ORR:150 West to NTS: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52 INEL to NTS: 31/13	East to ORR: 150 West to NTS: 43	998/902
	All to SRS	644	SRS to ORR: 161/65	East to ORR: 150 West to NTS: 43	998/902
	All to INEL	644	INEL to NTS: 161/65	East to ORR: 150 West to NTS: 43	998/902
HS/SRS	Geographic	East to SRS: 501 West to INEL: 143	INEL to HS: 36/15	East to SRS: 150 West to HS: 43	873/852
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	INEL to HS: 31/13	East to SRS: 150 West to HS: 43	868/850
	All to SRS	644	None	East to SRS: 150 West to HS: 43	837
HS/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51 INEL to HS: 36/15	East to ORR: 150 West to HS: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52 INEL to HS: 31/13	East to ORR: 150 West to HS: 43	998/902
	All to SRS	644	SRS to ORR: 161/65	East to ORR: 150 West to HS: 43	998/902
	All to INEL	644	INEL to HS: 161/65	East to ORR: 150 West to HS: 43	998/902

SRS = Savannah River Site, INEL = Idaho National Engineering Laboratory, HS = Hanford Site, ORR = Oak Ridge Reservation, NTS = Nevada Test Site

[a] *Truck/rail shipments, assuming that the truck casks used for interstate shipments are capable of carrying 4 times as much fuel, and rail casks 10 times as much fuel as the shipping cask received from the foreign research reactor.*

Rail routing is not covered by specific Department of Transportation and NRC regulations. Therefore, carriers would generally select the most direct route, which would serve to reduce travel time and radiation exposure consistent with track class and other rail service requirements.

NRC regulations concerning physical security and notification are set forth in 10 CFR 71 and 10 CFR 73, respectively. Carriers are required to submit proposed routes for spent nuclear fuel shipments to NRC for approval, and NRC publishes a public information circular that lists routes that have been evaluated and approved for specific spent nuclear fuel shipments (NRC, 1993).

Table 2-7 Shipment Summary for Centralization Alternatives

Spent Nuclear Fuel Site Option	Phase 1 Approach	Phase 1 Port-to-Site Shipments	Site-to-Site Shipments[a]	Phase 2 or Port-to-Final Site Shipments	Total Number of Shipments[a]
SRS	N/A - Single phase			837	837
INEL	N/A - Single phase			837	837
HS	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902
ORR	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902
NTS	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902

SRS = Savannah River Site, INEL = Idaho National Engineering Laboratory, HS = Hanford Site, ORR = Oak Ridge Reservation, NTS = Nevada Test Site

[a] *Truck/rail shipments assuming that the truck casks used for intersite shipments are capable of carrying 4 times as much fuel and rail casks 10 times as much fuel as the shipping cask received from the foreign research reactor due to consolidation.*

The HIGHWAY and INTERLINE computer codes are used to assist in route selection and estimations of exposed population (DOE, 1995c). The collective population risk, maximally exposed individual (MEI) risk, accident risk, accident consequence, and nonradiological risk assessments are performed using the RADTRAN and RISKIND computer codes established for shipment by both railroad and highway. Additional details of the treatment and analysis methodology used in the ground transportation assessment are given in Appendix E.

2.6.5 Activities and Alternatives at the Foreign Research Reactor Spent Nuclear Fuel Management Sites

The potential sites for receipt and management of foreign research reactor spent nuclear fuel are the same as those considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c), namely: the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

Since foreign research reactor spent nuclear fuel is part of the overall DOE spent nuclear fuel management program, the potential site-specific options are consistent with the site management alternatives considered in the Programmatic SNF&INEL Final EIS. The alternatives are: Decentralization and 1992/1993 Planning Basis (even distribution of foreign research reactor spent nuclear fuel between the Idaho National Engineering Laboratory and the Savannah River sites), Regionalization (distribution by fuel type and geography), and Centralization (all foreign research reactor spent nuclear fuel at the potential site).

As discussed earlier, the site-specific foreign research reactor spent nuclear fuel management options also depend on the availability of the management sites to implement the policy immediately. Of the five sites, only the Savannah River Site and the Idaho National Engineering Laboratory will be available in late 1995. The other three could become available at a later date when construction or refurbishment of appropriate facilities is completed. This constraint has resulted in the two-phased approach considered in some cases (see discussion in Section 2.6.4.1). For the purpose of the site impact analysis, the implementation of the policy was divided into two functional periods — the period during which receipt and management of foreign research reactor spent nuclear fuel is accomplished by using existing facilities (Phase 1), and the period during which new or refurbished facilities are used (Phase 2). For the environmental impact analysis, the first is characterized by operational activities only, while the second involves impacts from construction and operation activities.

Section 2.6.5.1 provides an overview of the storage technologies and descriptions of the storage facilities considered under the implementation alternatives of Management Alternatives 1 and 3. Section 2.6.5.2 provides a description of chemical separation, which is considered as an implementation alternative to storage.

The site-specific options selected for impact analysis are described separately in the sections devoted for each site (Sections 2.6.5.3.1 through 2.6.5.3.5).

2.6.5.1 Storage Technologies

The purpose of a spent nuclear fuel management facility is to provide an environment for the storage of spent nuclear fuel that protects the public, onsite workers, and the environment. The principal hazard presented by spent nuclear fuel is its inventory of radioactive elements that are the products of the reactions in a nuclear reactor. In addition, the fissionable uranium and plutonium remaining in the spent nuclear fuel has the potential of sustaining a fission chain reaction, which would generate additional radiation and fission products.

The management facility is designed to prevent the stored spent nuclear fuel from achieving a fission reaction (termed "criticality") and to isolate the radioactive materials within the spent nuclear fuel from the public and workers. Criticality is prevented by such methods as:

- maintaining a minimum separation distance between adjacent spent nuclear fuel elements;
- limiting the concentration of fissionable materials in each spent nuclear fuel storage container;
- installing neutron-absorbing materials between spent nuclear fuel elements; and
- controlling the presence and/or concentration of other materials that would enhance the ability of the stored spent nuclear fuel to become critical.

Protection of the public and workers from the radioactive materials within each spent nuclear fuel element is achieved by:

- enclosing or encapsulating the spent nuclear fuel so that any accidental release of radioactive material is retained;
- maintaining a benign chemical and thermal environment around the spent nuclear fuel so that its structural integrity is preserved;
- providing adequate shielding of the radiation emanating from the spent nuclear fuel so that dose rates outside the facility are lowered; and
- utilizing security barriers to isolate spent nuclear fuel from workers and public.

The technology for safely storing spent nuclear fuel (as defined by the above criteria) has been in use, in one form or another, for over 40 years in the nuclear industry. Spent nuclear fuel storage is generally characterized as either wet or dry, denoting whether the spent nuclear fuel elements reside in a water-filled pool or a dry atmosphere. Details of the concepts are provided in Appendix F, Section F.1.

The wet pool type of spent nuclear fuel storage is used at almost every water-cooled nuclear reactor in the world. There are currently more than 600 operating water-cooled power and research nuclear reactors, each with an individual storage pool. The pool design uses common materials (water and concrete) for spent nuclear fuel shielding, heat removal, and the confinement of any radioactive material that might be released from the spent nuclear fuel. An additional benefit is the ability to visually inspect spent nuclear fuel, since the water purity and clarity are maintained at a high level. Spacing, fissionable material limits, and in some cases, the use of neutron-absorbing material prevent criticality in a wet storage environment. The pool is enclosed in a suitably qualified structure or building. Construction of a wet storage facility involves excavating earth, backfilling, pouring concrete, setting piping, erecting a building around the pool, and installing piping, electrical systems, and heating, ventilating, and air conditioning systems. In many ways, a spent nuclear fuel storage pool is like a swimming pool, except its depth is greater and its concrete walls and floors are much thicker to provide for structural integrity. Wet storage facility designs include sophisticated methods of leak detection. To negate corrosion, the pool water purity and quality are carefully maintained and controlled.

Dry storage technology involves the encapsulation of spent nuclear fuel in a steel cylinder that may be placed in a concrete or massive steel cask or structure. The spent nuclear fuel is stored in racks within the cylinder or suspended from plates placed at variable distances in the cylinder, in either air, or inert atmosphere. Foreign research reactor spent nuclear fuel elements with suspect cladding integrity would be placed in sealed cans before they are placed in the cylinder (canning). Casks or structure materials, usually some form of concrete, steel, iron, or lead provide shielding and heat removal. Spacing, fissile material limits, and neutron absorbing materials are used to prevent criticality. Different forms of dry fuel storage have been used for over 40 years in the nuclear industry. Several nuclear power plants in the United States have licensed, built, and operated dry storage facilities during the last 7 years. NRC has reviewed and approved several manufacturers' designs for dry fuel storage of commercial spent nuclear fuel. Canada has been storing spent commercial nuclear power plant fuel in dry storage casks since 1975. Australia has been successfully storing its research reactor spent nuclear fuel since 1963 (Silver, 1993) in dry environment, and Japan has had 12 years of experience with dry storage of research reactor spent nuclear fuel (Shirai et al., 1991). The Savannah River Site has an ongoing developmental program on dry storage technology which would be used to implement this worldwide experience in the United States, and to finalize design parameters for a foreign research reactor spent nuclear fuel dry storage facility.

Dry storage methods are not as efficient in removing heat from the spent nuclear fuel as wet storage pools. Thus, as explained in Appendix F, this EIS assumes that high decay heat foreign research reactor spent nuclear fuel would initially be placed in wet storage. This would allow sufficient time for the spent nuclear fuel radioactive decay heat to decrease and not be a deciding factor in sizing a dry storage facility.

Dry storage facility construction involves the preparation and pouring of concrete foundations upon which the concrete or metal cask or building is then erected. Metal casks would be built away from the DOE site in a factory, since they involve thick metal fabrication techniques not used at DOE facilities. Concrete casks or buildings are constructed at the site using the same general principles (e.g., forms, rebar) as in nonnuclear concrete construction. Qualified concrete foundation pads are also poured for support bases of the casks.

Whether wet or dry storage were used, the facility would be designed to withstand natural phenomena such as earthquakes, floods, tornadoes, hurricanes, high and low temperatures, and wind generated missiles (branches, poles, etc.). The design would also include provisions to preclude sabotage or terrorist acts. Security requirements for a dry storage facility after the spent nuclear fuel has decayed to low levels of radioactivity and is no longer self-protecting would be met by the establishment of a Perimeter Intrusion Detection and Alarm System zone, which is the standard procedure for DOE. Each design has specific provisions for periodic inspection or surveillance, and must meet the highest quality standards associated with all safety requirements specified for nuclear facilities.

The current alternative types of storage technology are discussed and evaluated in detail in Appendix F. The basic categories are: wet (pool), dry concrete vault or building, dry concrete horizontal cask/module, dry concrete vertical cask/silo, dry metal vertical cask, hot cells, multi-purpose casks, and dry inground vertical holes. There are significant differences between these technologies in terms of construction, operations and maintenance costs and various design details. However, these differences do not result in any important variations in environmental impacts and consequences. With the exception of multi-purpose casks (which are still under development), all of these technologies have proven records of successful safe operation while storing spent nuclear fuel. Appendix F provides detailed descriptions concerning generic dry and wet storage facilities for foreign research reactor spent nuclear fuel. Brief descriptions of both wet and dry storage facilities are provided in the following sections.

2.6.5.1.1 Description of Dry Storage Facilities

Spent Nuclear Fuel Storage Using a Modular Dry Vault:

An aboveground dry vault is a self-contained concrete structure that would allow for dry spent nuclear fuel handling and storage. This design represents an integrated spent nuclear fuel storage approach and would consist of four major components: a receiving/loading/inspection area, spent nuclear fuel storage canisters, a shielded canister handling machine, and a modular array for storing the spent nuclear fuel storage canisters. Figure 2-7 displays an illustration of a typical modular dry vault storage facility. The receiving area would use a wet pool for unloading the casks and for short-term (1 to 3 years) storage of foreign research reactor spent nuclear fuel elements with a heat load exceeding 40 Watts per element. The vault would consist of several modular units, and each unit could provide storage for hundreds of spent nuclear fuel assemblies. The vault itself would contain a charge/discharge bay with a spent nuclear fuel handling machine above a floor containing steel tubes that house the (removable) spent nuclear fuel canisters. The bay would be shielded from the stored spent nuclear fuel by the thick concrete floor and shield plugs inserted into the top of the steel storage tubes. The steel tubes would serve as secondary containment for the foreign research reactor spent nuclear fuel and would descend into an open storage

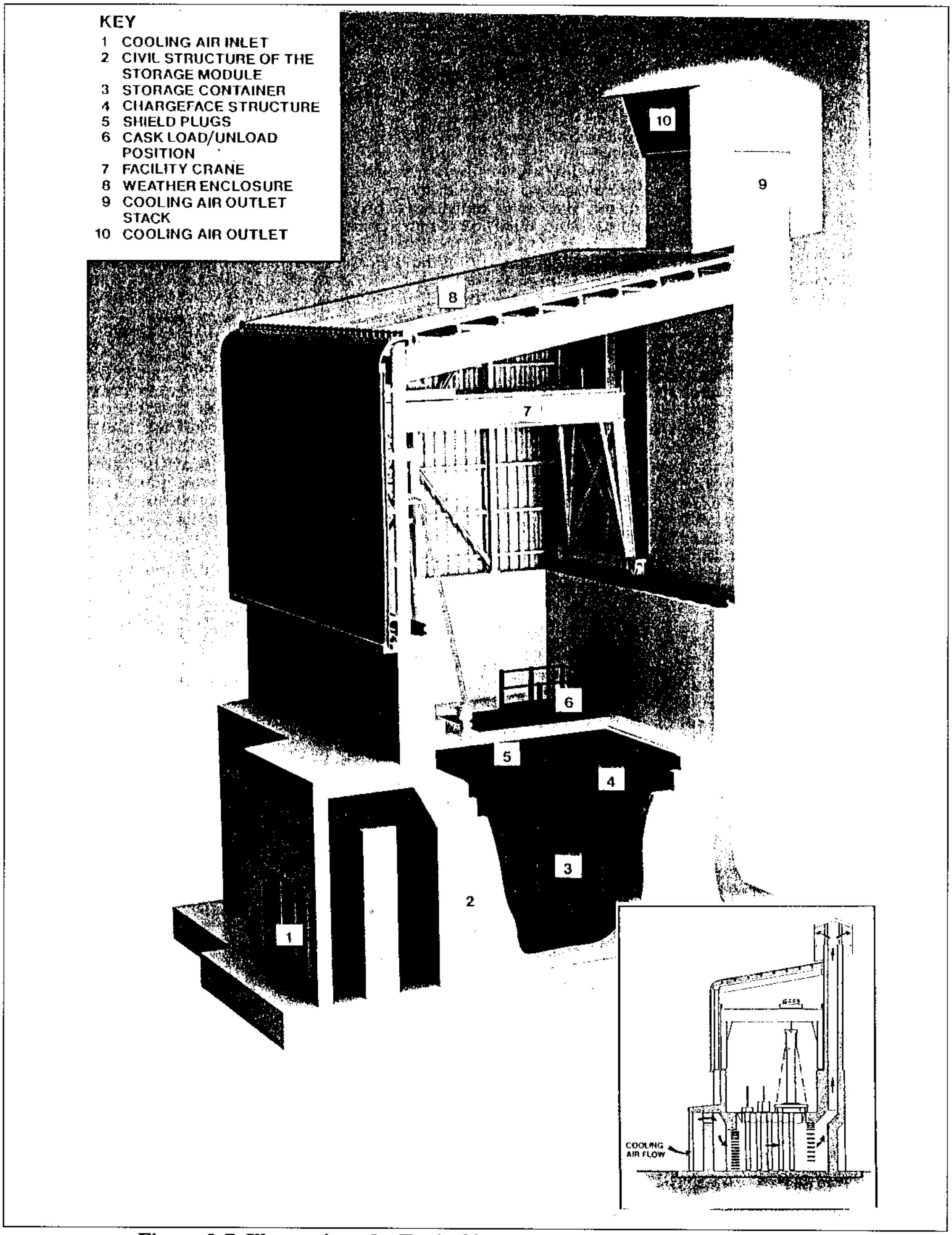


Figure 2-7 Illustration of a Typical Modular Dry Vault Storage Facility

Table 2-8 Summary of Modular Dry Vault Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

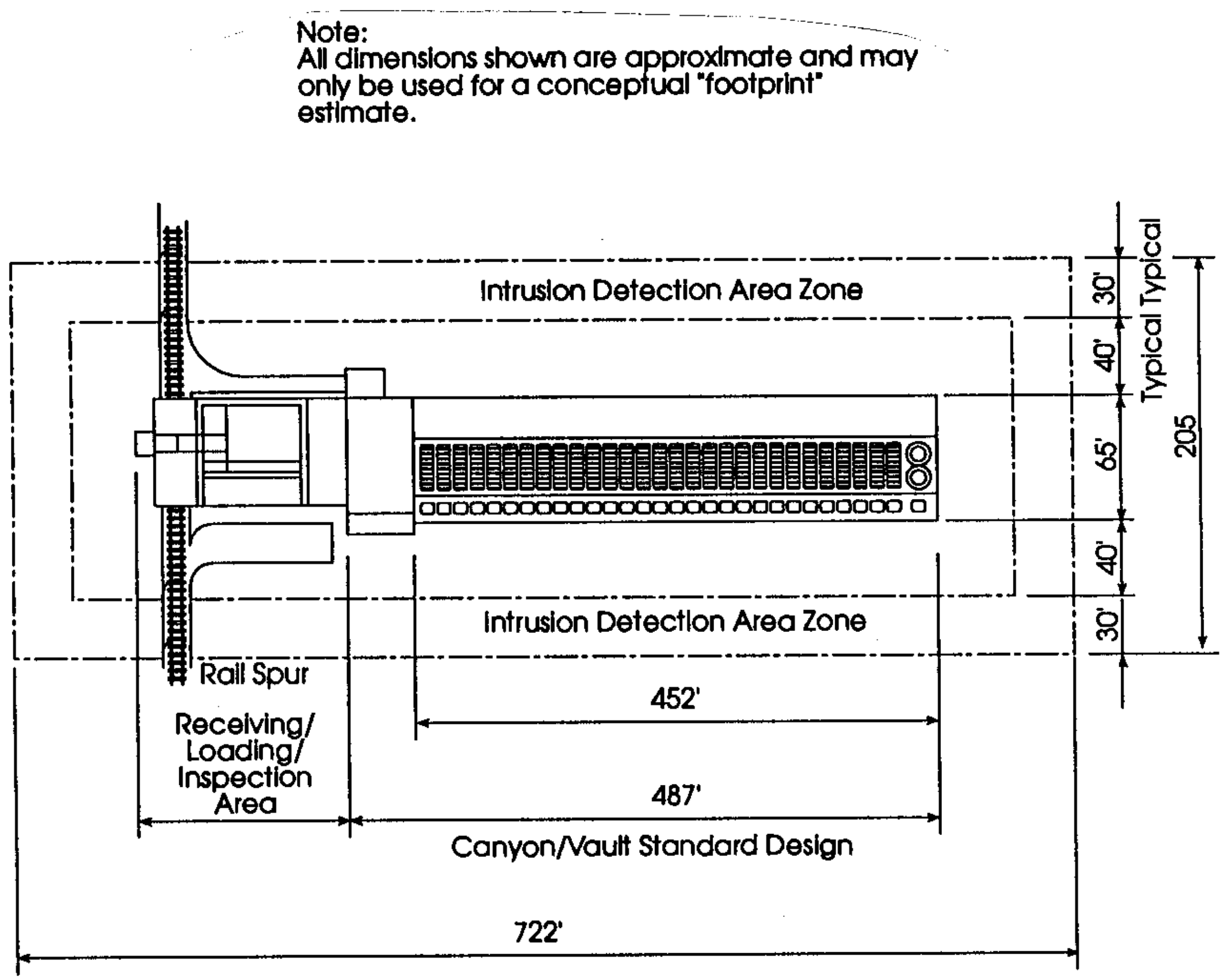
Construction Phase:	
Disturbed Land Area	3.7 ha (9 acres)
Facility: size (area) concrete steel	 5,000 m^2 (54,000 ft^2) 21,800 m^3 (28,500 yd^3) 5,200 metric tons (5,750 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	835,000 l (221,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	190/yr (average), 234/yr (peak)
Duration (years)	4 years for construction, 1.5 years for design
Capital Cost	$370 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/year) during receipt 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams Solid Low-Level Waste Waste Water	 22 m^3/yr (780 ft^3/yr) during receipt 1 m^3/yr (35 ft^3/yr) thereafter 1.59 million l/yr (420,000 gal/yr) during receipt 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt 8 thereafter
Annual Operating Cost	$15.6 million during handling, $0.6 million during storage[b]

[a] *Staging facility parameters are based upon the regionalized, small wet pool (Dahlke, et al., 1994)*

[b] *Cost estimates are in 1993 dollars (EG&G, 1993)*

area. Large, labyrinth air supply ducts and discharge chimneys would permit natural convection cooling of the steel spent nuclear fuel storage tubes, while the perimeter concrete walls would provide for shielding. The design would allow for expansion by adding additional units of arrays to the end of the vault or by construction of another vault. The vault facility would also include a receiving and loading bay that would allow handling of shielded transportation casks and unloading of the foreign research reactor spent nuclear fuel into the short-term wet storage pool. The receiving bay provides for spent nuclear fuel inspection, canning as required and could be used for spent nuclear fuel characterization with additional equipment and modifications. Although it is not expected that the physical condition of the foreign research reactor spent nuclear fuel elements would require extensive canning, the capability of canning the entire foreign research reactor spent nuclear fuel inventory would be provided by the design. Table 2-8 summarizes modular dry vault storage parameters for foreign research reactor spent nuclear fuel storage.

In operation, the transportation cask would be lifted by a crane and placed in the unloading area of the small wet pool. The fuel elements would be removed underwater, examined, and if the heat generation rate is below 40 Watts per element, the spent nuclear fuel would be placed within the transfer canister. The transfer canister would be subsequently drained, dried, and seal-welded. The handling machine then would place the spent nuclear fuel inside of the spent nuclear fuel storage canister, and would transport the loaded canister to the storage tubes. The handling machine would include radiation shielding. Heat dissipation would be accomplished by natural convection from the surfaces of the handling machine and canister. Decay heat would be dissipated by natural convection: air would enter through inlet ducts at the bottom of the vault module, pass around the outside of the steel storage tubes containing the spent nuclear



10-40 W/Element Scenario
25,000 FRR spent fuel element modular array (Nominal) Storage req. (29 modules with 44 tubes per module - 20 spent fuel elements per tube)
• Passive cooling • Monitored storage canisters • Retrievable for periodic inspection • Self-contained facility for staging and storage of spent fuel elements

Figure 2-8 Layout of a Modular Dry Vault Storage Facility for Foreign Research Reactor Spent Nuclear Fuel (10 Watt to 40 Watt Element Basis)

fuel canisters, and exit through outlet ducts at the top of the module. Therefore, the vault would be a complete, integrated facility with all of the required capabilities for foreign research reactor spent nuclear fuel handling and storage.

The vault facility would store spent nuclear fuel in canisters that are approximately 40.6 cm (16 in) in diameter by 4.6 m (15 ft) long. As currently envisioned, foreign research reactor spent nuclear fuel would be stored within the canister in 5 levels with 4 elements per level, for a total of 20 spent nuclear fuel elements per canister (MTR-type design). The vault design would allow for 36 to 44 canisters per array unit, depending upon the decay heat of the spent nuclear fuel and a cladding temperature limit nominally 175°C (347°F) for aluminum-cladding with an air inlet temperature of 49°C (120.2°F). Thus, the number of vault units/arrays required for the storage of elements having a decay heat between 10 Watts and 40 Watts per element would be 27.

Most of the foreign research reactor spent nuclear fuel is expected to have decay heats between 10 Watts and 40 Watts per element. For "cold" fuel (less than 10 Watts per element), potentially more than 44 spent nuclear fuel canisters could be placed per vault unit. However, this would require a customized design, which could unnecessarily increase costs and implementation time. Figure 2-8 displays the layout of the modular dry vault storage facility (10 Watt to 40 Watt element basis).

Criticality concerns would be addressed primarily by the tube spacing in the vault. Borated concrete could also be used. For foreign research reactor spent nuclear fuel, criticality would not be expected to be a significant concern because a considerable fraction of the fissile uranium would have been consumed, and neutron-absorbing fission products would be present.

This vault design, without a pool, has been licensed by NRC for the Fort St. Vrain nuclear power plant site. It represents a complete, stand-alone facility that could be dedicated to foreign research reactor spent nuclear fuel without requiring the utilization of any other facilities at the host site. Cask handling, spent nuclear fuel transfer to a canister, and spent nuclear fuel storage could be accomplished within the facility. Additional facilities or modifications to the inspection area, including a pool, would be required for foreign research reactor spent nuclear fuel characterization.

The cost to construct a modular dry vault storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the vault storage area is estimated to be $370 million. The annual operating cost for this facility is estimated to be $15.6 million during the period of handling and transfers of the spent nuclear fuel and $0.6 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

Spent Nuclear Fuel Storage Using Dry Casks:

Dry cask storage would include the use of concrete casks, both vertical and horizontal versions, metal casks, and multipurpose casks and would consist of the following components:

- A staging facility for cask receipt and unloading, and for loading foreign research reactor spent nuclear fuel into the dry storage casks. The staging facility would have a wet pool for unloading the casks and for short-term (1 to 3 years) storage of spent nuclear fuel with a heat load exceeding 40 Watts per element. This facility would include capabilities for drying the spent nuclear fuel/canister, inserting the spent nuclear fuel/canister with helium or nitrogen, and welding the storage canister closed.

- An inspection/characterization facility, for examining spent nuclear fuel integrity and canning leaking spent nuclear fuel as required. This may be incorporated into the staging facility (as an inspection cell) or be immediately adjacent to it. Although it is not expected that the physical condition of the foreign research reactor spent nuclear fuel elements would require extensive canning, the capability of canning the entire foreign research reactor spent nuclear fuel inventory would be provided by design.

- A dry storage cask (usually concrete). This would provide for the shielding and the structural stability of the spent nuclear fuel storage. The Multi-purpose Canister undergoing development could also be used (see Appendix F, Section F.1).

- A transfer mechanism, such as a dedicated truck/trailer combination with a ram for horizontal modules or a crane for vertical modules.

- A separate spent nuclear fuel canister may or may not be used. If used, it would typically be approximately 4.6 m (15 ft) long and 1.7 m (5.5 ft) in diameter, and would weigh approximately 33 metric tons (36 tons).

The dry cask approach would require the staging facility to receive and inspect the spent nuclear fuel shipment. The transportation cask would be unloaded in a small wet pool within the facility. Subsequently, spent nuclear fuel would be loaded into the dry cask (or spent nuclear fuel canister for the horizontal cask), and the cask would be placed on a concrete slab located outdoors. The horizontal approach would use a dry spent nuclear fuel transfer canister for containing the spent nuclear fuel. This would be placed within a shielded transfer cask and moved to the outside modular storage facility. A hydraulic ram would insert the transfer canister inside the horizontal storage module, followed by sealing with a shield plug. Thus, dry cask storage would always rely on the use of another facility.

Dry storage casks would be designed to withstand normal loads and design basis accident effects, such as earthquakes, tornadoes, and floods. Concrete would provide radiation shielding for gamma rays and neutrons. Natural air circulation would dissipate the heat; air would enter through inlet vents near the bottom of the cask, pass around the spent nuclear fuel canister, and exit near the top. Screens and grills would keep birds and animals out of the cooling duct area.

Some of the potential management sites have facilities which could be used for cask receipt and unloading and spent nuclear fuel inspection and transfer to storage. Utilization of these facilities would be considered.

The application of dry cask storage technology to foreign research reactor spent nuclear fuel would depend upon the heat load. Horizontal casks are anticipated to be slightly more restrictive than the vertical casks with respect to the heat load and are thus the focus of discussion. The standard design for a horizontal fuel canister would provide for 24 or 52 sleeves (i.e., pressurized water reactor or boiling water reactor spent nuclear fuel, respectively), each about 4.6 m (15 ft) long. As with the vault approach, it would be conservatively assumed that each sleeve contains 5 foreign research reactor spent nuclear fuel elements (i.e., in layers) within a basket or can arrangement for maintaining spacing and retrievability. Also, as with the vault approach, the number of dry storage casks would depend upon the decay heat of the spent nuclear fuel and a cladding temperature limit [nominally, 175°C (347°F) for aluminum-cladding with an air inlet temperature of 49°C (120.2°F)]. The 24-sleeve design would allow for a maximum of 120 elements for foreign research reactor spent nuclear fuel with 40 Watts to 80 Watts per element of decay heat, while the 52-sleeve design would provide for a minimum of 260 elements per dry storage cask with 10 Watts to 40 Watts per element. Thus, based on the total number of elements for which the facilities are sized, the number of casks required would be:

- ninety-four casks, predicated upon a 3-year cooldown period (i.e., less than 40 Watts per element). Note that this value is conservative and corresponds to a maximum of around 40 percent of the NRC-licensed heat loads per cask. Again, most foreign research reactor spent nuclear fuel is expected to have decay heats between 10 Watts and 40 Watts per element. Initially, foreign research reactor spent nuclear fuel with higher heat loads could be unsuitable for the dry storage cask pending detailed heat transfer analysis and a final determination of limiting fuel storage temperature for aluminum-based and TRIGA-type spent nuclear fuel. However, the relatively high decay heat spent nuclear fuel represents such a small percentage of the currently identified foreign research reactor spent nuclear fuel that its impact would be small, such that after 3 years of wet storage, it would all be below a heat output of 40 Watts per element.

Figure 2-9 displays the general layout for the dry cask storage facility predicated upon a horizontal cask design. Table 2-9 summarizes dry cask storage parameters.

Dry storage cask technology would require a separate staging facility for foreign research reactor spent nuclear fuel unloading, canning, and storage cask loading, and transportation cask maintenance. This facility would have the following operational areas:

- Transportation Cask Handling: this incorporates transportation cask maintenance, truck/railcar unloading, decontamination/washdown, radioactive material control, and cask sampling/flushing/degassing.

- A Small Wet Storage Pool: for fuel transfer and short-term storage.

- Spent Nuclear Fuel Unit Handling: fuel removal, decontamination, fuel drying, fuel canning, inserting with helium, and thermal measurements.

- Spent Nuclear Fuel Unit Transfer: this constitutes placement of the spent nuclear fuel into the cask or canister, followed by sealing.

- Radwaste Treatment: this includes collection, treatment, and preparation for disposal of contaminated effluents, and radioactive waste treatment and solidification.

- Heating, Ventilating, and Air Conditioning: this represents the component of the facility that helps ensure that contamination of workers and the environment is avoided.

The inspection/characterization facility would include a shielded dry hot cell for spent nuclear fuel analysis and examination, and canning of leaking spent nuclear fuel. All equipment and instrumentation within the cells would be remotely operated. The facility would be maintained under negative pressure with exhaust through high-efficiency particulate air filters to mitigate the environmental effects of any radionuclide releases. This facility is normally immediately adjacent to, or within, the staging facility.

Dry cask storage is unique among the three storage technologies because of its ability to be operationally integrated with existing facilities, which allows for faster implementation as compared to the other two storage technologies. Several management sites have facilities with spent nuclear fuel handling capabilities similar to the requirements of the staging facility. Potential examples include the Receiving

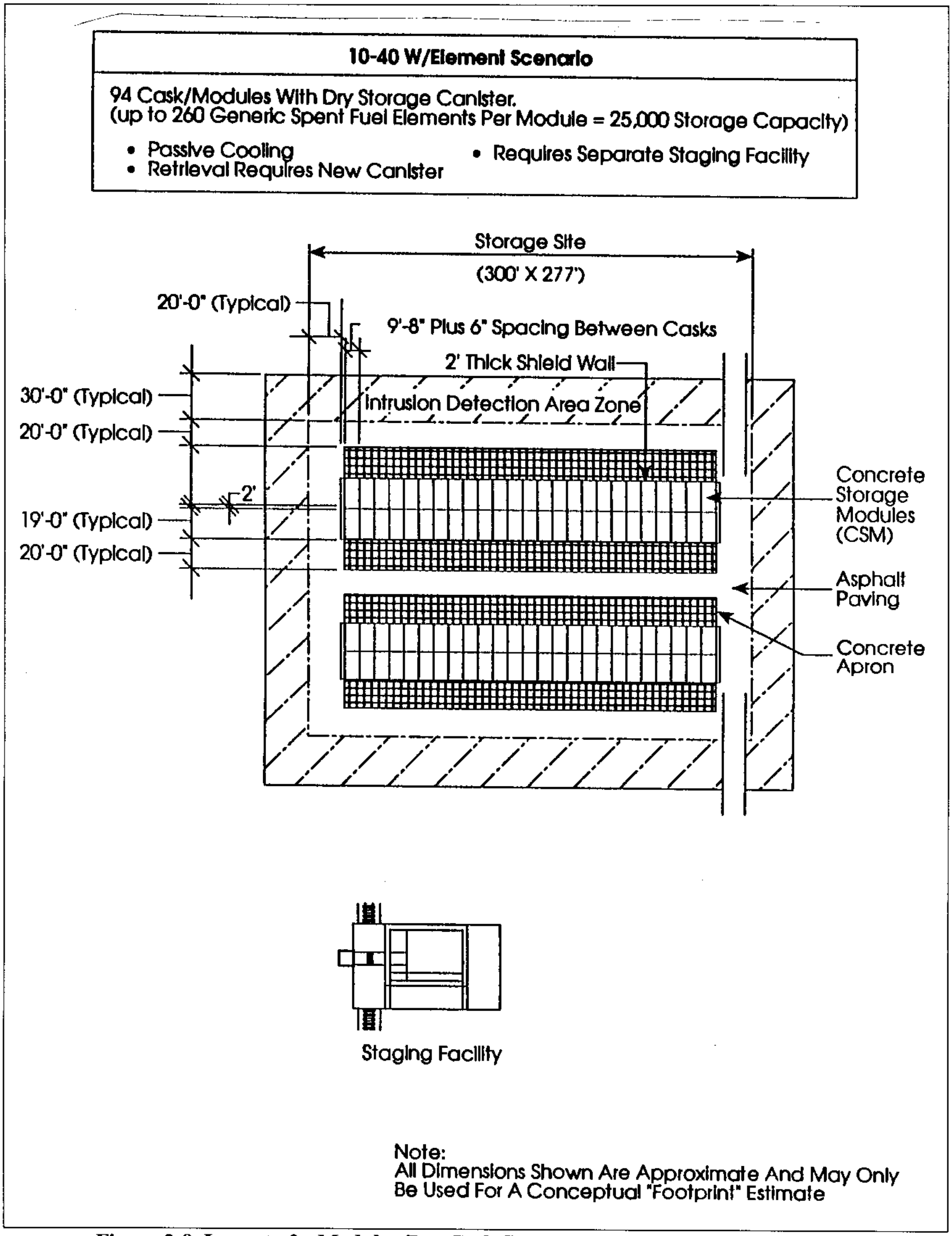


Figure 2-9 Layout of a Modular Dry Cask Storage Facility for Foreign Research Reactor Spent Nuclear Fuel (10 Watt to 40 Watt Element Basis)

Table 2-9 Summary of Dry Cask Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3 ha (7.7 acres)
Facility: size (area) concrete steel	 2,200 m^2 (24,000 ft^2) 17,500 m^3 (22,900 yd^3) 4,500 metric tons (5,000 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	810,000 l (214,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	50/yr for staging facility 50 per 24 cask array, 1 array per year
Duration (years)	5.5 for staging facility 4 years for construction, 1.5 years for design
Capital Cost	\$366 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/year) during receipt 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams Solid Low-Level Waste Waste Water	 16 m^3/yr (565 ft^3/yr) during receipt 1 m^3/yr (35 ft^3/yr) thereafter 1.58 million l/yr (412,000 gal/yr) during receipt 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt 8 thereafter
Annual Operating Cost	\$17.3 million during handling, \$0.3 million during storage[b]

[a] *Staging facility parameters are based upon the regionalized, small wet pool (Dahlke et al., 1994)*

[b] *Cost estimates are in 1993 dollars (EG&G, 1993)*

Basin for Offsite Fuels (RBOF) at the Savannah River Site and the CPP-666 storage pool area at the Idaho National Engineering Laboratory. For dry cask storage, the spent nuclear fuel would be shipped to the existing facility and unloaded from the transportation cask. The spent nuclear fuel would be inspected, canned if identified as a leaking element, and placed inside the storage canister. Spent nuclear fuel elements with heat loads exceeding 40 Watts per element would be stored in the existing facility to allow cooldown prior to cask storage. After filling, the canister would be sealed and placed inside the storage cask. The only new construction required would be the concrete storage pad (for vertical casks) or the concrete storage modules (for horizontal casks).

The cost to construct a dry cask storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the cask storage area is estimated to be \$366 million. The annual operating cost for this facility is estimated to be \$17.3 million during the period of handling and transfers of the spent nuclear fuel and \$0.3 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

2.6.5.1.2 Description of Wet Storage Facilities

A wet storage facility consists of a spent nuclear fuel storage area and support areas (Dahlke et al., 1994). The spent nuclear fuel management area would provide for the receipt of cask transport vehicles, cask unloading and decontamination, and spent nuclear fuel handling, transfer, and storage. Support areas would provide for the equipment necessary to maintain and operate the storage area (e.g., heating, ventilating, and air conditioning; water treatment; and waste management). The general layout of a wet storage facility is presented in Figure 2-10. The wet storage facility would be constructed as a safety class structure that meets all current nuclear regulations to withstand natural events such as seismic activity, tornadoes, and floods, as well as aircraft impact. Systems supporting the operation of the spent nuclear fuel management facility would also be required to meet these safety requirements. The facility would be equipped with a 118-metric ton (130-ton) overhead crane and a 9-metric ton (10-ton) spent nuclear fuel handling crane. Figure 2-11 displays a schematic of the facility, and Table 2-10 summarizes wet storage parameters for foreign research reactor spent nuclear fuel handling and storage.

Each cask transport vehicle would enter the facility through one of two bays where it would be monitored and washed to remove transportation dust. When the external surfaces are cleaned, the cask would be placed into a decontamination room where the cask would be prepared as needed to facilitate underwater unloading. The cask would then be placed in an unloading pool. The cask receiving area can accept two simultaneous shipments on 3 m by 24.4 m (10 ft by 80 ft) trucks or railcars, and casks weighing up to 114.3 metric tons (126 tons) each with a total individual cask and transport vehicle weight of 176 metric tons (195 tons). There are two unloading pools [6.1 m long and wide by 11.0 m deep (21 ft long and wide by 36 ft deep) and 6.4 m long by 5.8 m wide by 13.4 m deep (21 ft long by 19 ft wide by 44 ft deep)] and two decontamination rooms. Prior to being placed in one of the two storage pools, each fuel element would be checked to ensure that it is properly configured for direct transfer to the fuel storage pool buckets. If not, it would be transferred to the fuel cutting/canning pool [10.4 m long by 5.8 m wide by 9.4 m deep (34 ft long by 19 ft wide by 31 ft deep)] where it would be prepared for transfer to the storage pool buckets.

If cask measurements indicated that the spent nuclear fuel might be leaking, the spent nuclear fuel would be transferred to the isolation pool [3.7 m long by 3.0 m wide by 9.4 m deep (12 ft long by 10 ft wide by 31 ft deep)] for sipping. Sipping is a methodology for determining leaking spent nuclear fuel. This pool would be equipped so that wet sipping, dry sipping, or vacuum sipping of the suspect spent nuclear fuel element could be performed. An identified leaking spent nuclear fuel element would then be transferred to the cutting/canning pool where it would be canned before transfer to one of the storage pools. If it was not found to be leaking, it would be transferred directly to a storage pool.

All six pools in this facility (two unloading, two storage, one cutting/canning, and one leak check/isolation) would be hydraulically connected by a transfer channel/pool which would be 6.1 m long by 3.3 m wide by 9.4 m deep (20 ft long by 11 ft wide by 31 ft deep). Gates between this transfer channel and each pool would allow for hydraulic watertight isolation of the other pools. All pools and channels would be constructed of concrete with stainless steel floors and liners. Pool water leak detection and collection systems in accordance with NRC Regulatory Guide 1.13 (NRC, 1975) and American National Standards Institute, Standard N305-1975 (ANSI, 1975) would be provided for the pools.

Each of the two storage pools would be 16.5 m long by 10.4 m wide by 9.4 m deep (54 ft long by 34 ft wide by 31 ft deep), and each would contain 40 stainless steel storage racks. This would provide 1,000 storage holes with a 20 cm (8 in) spacing maintained between adjacent holes. The 20 cm (8 in) space provides neutron isolation between adjacent spent nuclear fuel elements, and would ensure criticality safety. Each rack would be 2.0 m square and 3.2 m high (6.7 ft square and 10.5 ft high) and consist of a

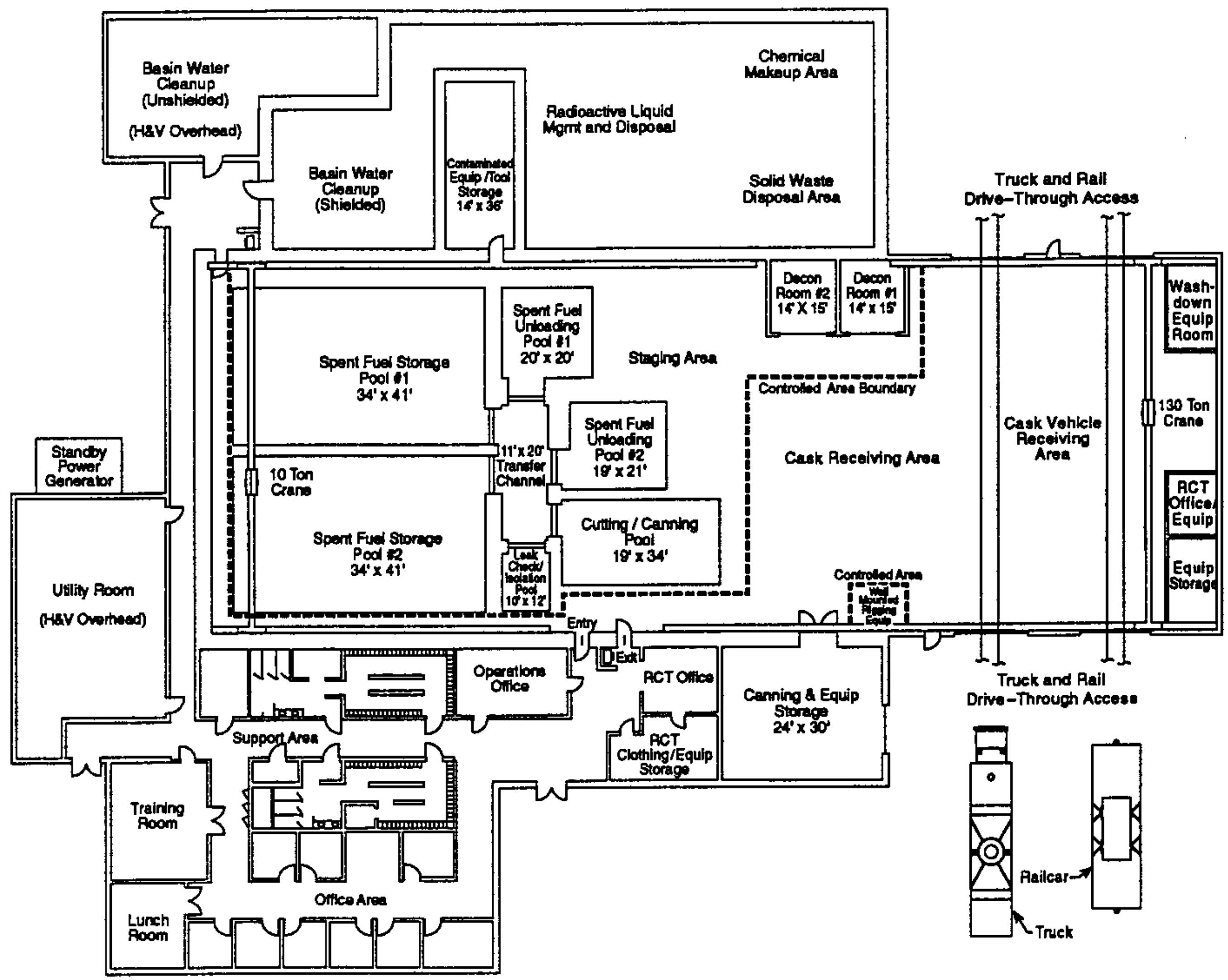


Figure 2-10 Generic Wet Storage Facility for Foreign Research Reactor Spent Nuclear Fuel

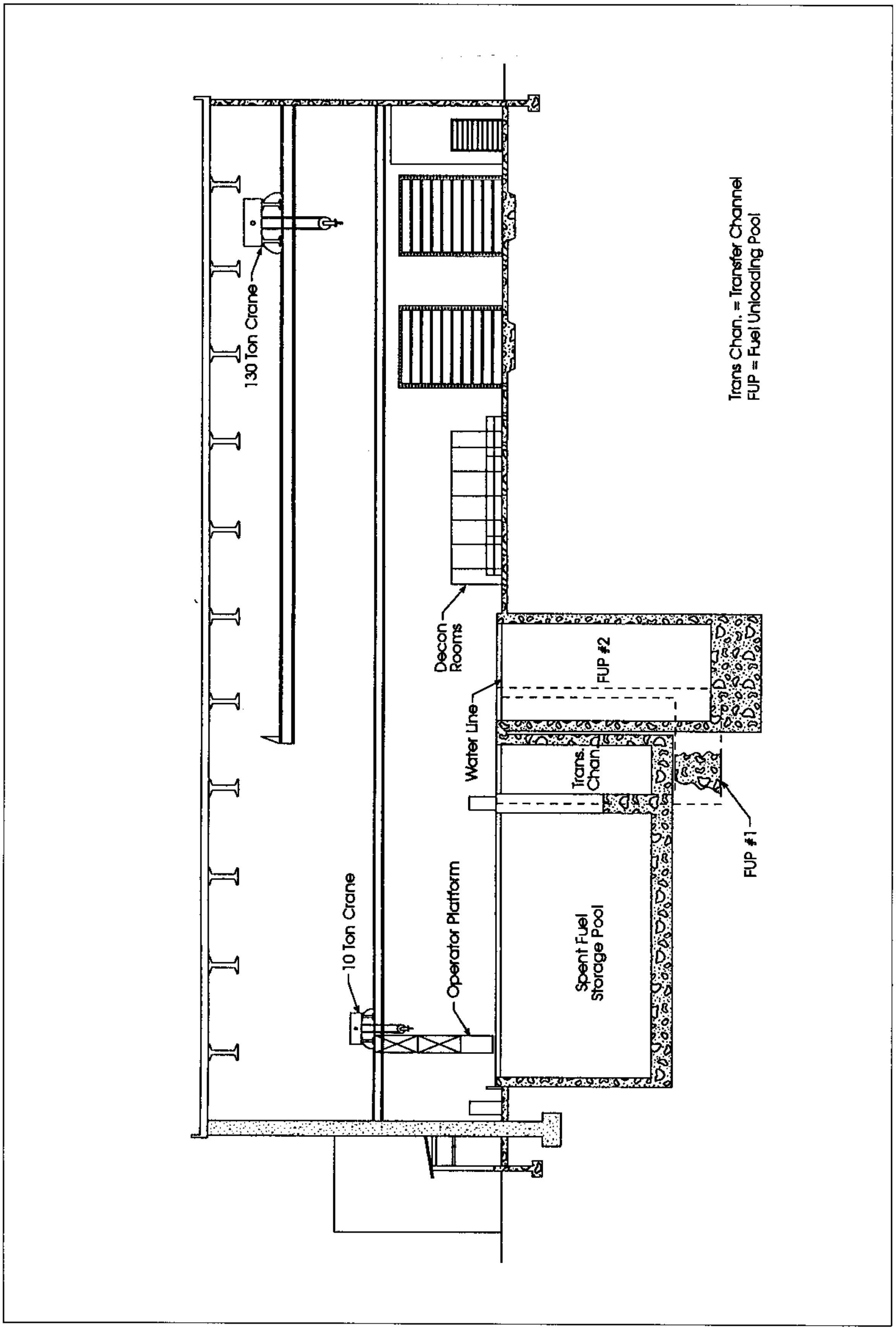


Figure 2-11 Schematic of a Wet Storage Facility for Foreign Research Reactor Spent Nuclear Fuel

Table 2-10 Summary of Wet Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel

Construction Phase:	
Disturbed Land Area	2.8 ha (7 acres)
Facility: size (area) concrete steel	 3,800 m^2 (41,000 ft^2) 12,400 m^3 (16,260 yd^3) 3,100 metric tons (3,443 tons)
Soil Moved	18,000 m^3 (24,000 yd^3)
Equipment Fuel	600,000 l (159,000 gal)
Construction Debris/Waste	2,600 m^3 (10,300 yd^3)
Work Force	157/yr (average), 184 peak
Duration (years)	4 years for construction, 1.5 years for design
Cost	\$449 million[a,b]
Operation Phase:	
Electricity	1,000 - 1,500 MW-hr/yr
Water (liters)	2.7 million l/yr (720,000 gal/yr) during receipt 1.5 million l/yr (409,000 gal/yr) thereafter
Waste Streams:	
High-Level Waste	none
TRU	none
Solid Low-Level Waste	16 m^3/yr (580 ft^3/yr)
Waste Water	1.59 million l/yr (420,000 gal/yr) during receipt 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30
Annual Cost	\$23.3 million during handling, \$3.5 million during storage[b]

[a] *Cost estimates are in 1993 dollars (EG&G, 1993)*

[b] *The cost may include duplicate facilities and equipment present in both the staging and the rest of the wet storage facility.*

5 by 5 array of 25 spent nuclear fuel positions. A hinged lid would be above each of these spent nuclear fuel positions. Spent nuclear fuel elements would be stored in the racks so that at least 30 cm (12 in) of rack would protrude above the top of the fuel. Each position in the rack can hold up to three storage buckets, which would be stacked vertically on top of each other. The bucket, made up of 3.175 mm (0.125 in) thick stainless steel, would be fitted with ceramic spaces to prevent galvanic corrosion, and could store either two or four spent nuclear fuel elements, depending on the specific fuel design. This would provide a total capacity of approximately 12,000 elements for each storage pool.

The heating, ventilation, and air conditioning system for the wet storage facility would include a room of air supply equipment and a room for air exhaust equipment with separate filtering and monitoring. All exhaust air would be directed through pre-filters, high-efficiency particulate air filters, radiation monitors, filter fire protection components, and heat recovery coils before it would exhaust to the atmosphere.

The wet storage facility's water treatment system would consist of redundant pumps, piping, filters, deionizers and microorganism control systems. A heat removal system would be sized to maintain the bulk water temperature to acceptable levels. The system's filters and deionizers would include anion and cation exchangers that maintain water chemistry and remove radionuclides from the pool water.

The staff required to operate the wet storage facility would be a maximum of 30 when 24-hour-a-day fuel loading was being performed.

No high activity solid radioactive waste would be generated by the wet storage facility (equivalent to Class B or C low-level waste) over the life of the facility. Low-level solid radioactive waste that would be generated over the life of the facility would be about 640 m^3 (22,600 ft^3). Nonradioactive solid waste generated over the facility's life would be about 300 m^3 (10,600 ft^3). All ventilation air would pass through roughing and high-efficiency particulate air filters prior to exhaust. No nonradioactive, hazardous air emissions would be generated by this facility.

The cost to construct a wet storage facility with a staging area sufficient to unload, characterize, can, and transfer the spent nuclear fuel to the storage area is estimated to be $449 million. This cost may include some duplicate facilities and equipment present in both the staging facility and the rest of the wet storage facility which were costed separately. The annual operating cost for this facility is estimated to be $23.3 million during the period of handling the spent nuclear fuel and $3.5 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993).

2.6.5.2 Chemical Separation

Chemical separation involves separating the fissile material in the spent nuclear fuel from the other material (i.e., cladding material, fission products, etc.). Uranium and plutonium isotopes constitute the fissile materials; and with foreign research reactor spent nuclear fuel, relatively little plutonium and actinide elements are produced because the ^{238}U precursor is present in relatively small quantities. Waste materials would be mainly fission products (radioactive species such as cesium and strontium) in the form of liquid raffinates, low-level radioactive wastes, mixed radioactive/chemical wastes, waste acids, chelating and complexing agents, and organic solvents. The highly radioactive nature of fission products would require that the chemical separation activities be performed. Plutonium can be handled in facilities without radiation shielding, although these materials would still have to be handled under special procedures and precautions due to their radioactive, fissile nature. The other waste forms would require specialized handling, including volume reduction in some cases, to allow for safe storage and disposal.

Aqueous chemical methods are the only processing method applied on a large scale. All existing plants use an extraction process, which has been used for some 40 years. The spent nuclear fuel is initially dissolved in an acid and contacted with an organic solvent containing an extractant, such as tributylphosphate. The uranium and plutonium form a complex with the tributylphosphate and transfer to the organic phase. The cladding and waste materials remain in the aqueous phase, which is termed high-level waste. The uranium and plutonium are subsequently recovered by contact of the organic phase with weak acidic solutions. Vitrification of the high-level waste is the preferred waste management approach.

As discussed in Section 2.2.2.6 chemical separation is not a preferred technology for managing spent nuclear fuel in the United States.

Processing facilities exist at several DOE and foreign sites. The main domestic facilities are located at the Savannah River Site and Idaho National Engineering Laboratory. The main foreign facilities are in France and the United Kingdom.

Savannah River Site Facilities

At the Savannah River Site, two facilities are available to chemically separate the foreign research reactor spent nuclear fuel. These facilities are the F- and H-Canyons. The F- and H-Canyon facilities are nearly identical structures that use similar radiochemical processes for the separation and recovery of plutonium, neptunium, and uranium isotopes. The F-Canyon primarily recovered ^{239}Pu and ^{238}U from irradiated natural or depleted uranium, and the H-Canyon primarily recovered ^{238}Pu, ^{237}Np, and ^{235}U from irradiated reactor fuels and targets. The following paragraphs apply to both canyons unless noted.

The F- and H-Canyons are reinforced concrete structures, 255 m long by 37 m wide, and 20 m high (836.6 ft by 308 ft by 121.4 ft). They are named for the two areas ("canyons") in each structure that house the large equipment (tanks, process vessels, evaporators, etc.) used in the chemical separations processes performed in each facility. These areas are 170 m long by an average of 6 m narrow and 20 m deep (557.7 ft by 19.7 ft by 65.6 ft). The two canyons are parallel and open from floor to roof. A center section, which has four floors or levels, separates the canyons. The center section contains office space, the control room for all facility operations, chemical feed systems, and support equipment such as ventilation fans. Processing operations involving high radiation levels (dissolution, fission product separation, and high-level radioactive waste evaporation) occur in the "hot" canyon, which has thick concrete walls to shield people outside the facility and in the center section from radiation. The final steps of the chemical separations process, which generally involve lower radiation levels, occur in the "warm" canyon. Figure 2-12 shows the layout of F-Canyon.

Services typical for a large industrial facility are required to support the canyon operations. Such services include steam and cooling water for process vessels and a ventilation system.

A separate ventilation system serves portions of the facility, such as the hot and warm canyons, that contain the radioactive process equipment. This system ensures that the air pressure in such areas is below the pressure of the air outside the facility and the area occupied by workers. This design helps prevent the release of radioactive material outside the facility by ensuring that air always flows from the outside of the facility to the inside of the process areas. Air in the process areas is exhausted from the facility through a large filter that removes 99.5 percent of any airborne radioactive material from the air. A 61-meter-tall (200-ft) stack behind each canyon discharges this air to the atmosphere. This stack is the pathway for airborne emissions associated with the normal operation of the canyons.

Even though DOE has maintained the chemical separation facilities since their construction, they contain equipment and systems that have become degraded because of their age and changes in mission. In some cases the degraded condition of equipment can pose operational limitations. For example, at one time the H-Canyon contained equipment that provided the capability to dissolve not only aluminum-clad reactor fuel but also fuel clad in stainless steel. The electrolytic dissolver used for this purpose is no longer functional and has been abandoned in place.

Because of the ages of the facilities, they do not satisfy all current DOE requirements for the design and construction of nuclear facilities. For example, the canyons and associated B-Line facilities were built (during the Cold War when a primary concern was the potential for an attack) to resist a large external blast. The blast-resistant features of the canyons also make them resistant to such external natural phenomena as tornadoes and earthquakes. However, the canyons were not designed to withstand a severe earthquake (defined as producing a lateral ground acceleration that is 20 percent that of gravity or 0.2 g), as they would be if DOE were to build them today.

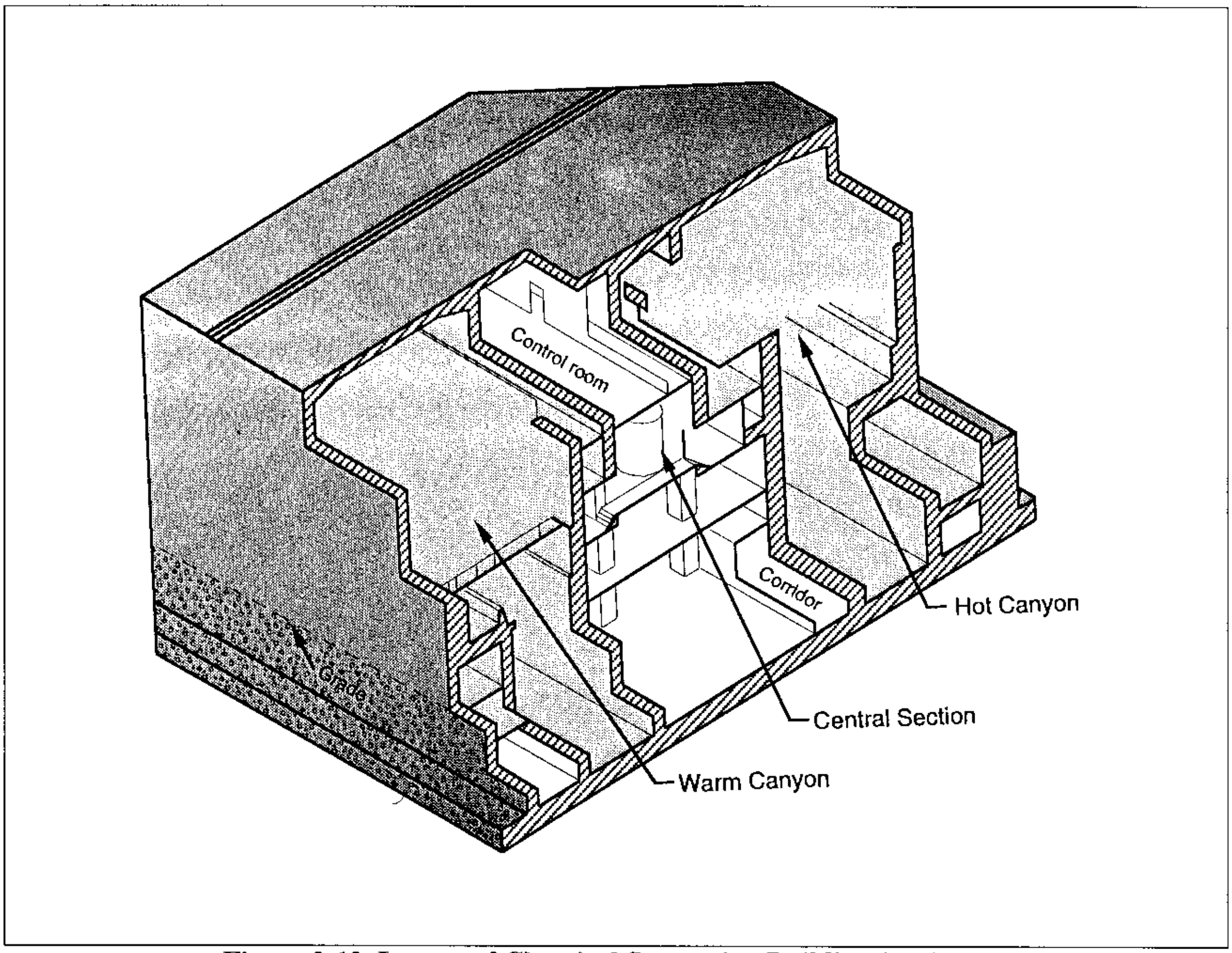


Figure 2-12 Layout of Chemical Separation Building Sections at Savannah River Site

The continued use of these facilities to chemically separate nuclear materials is an important factor for DOE consideration. Because the facilities do not meet current design and construction requirements, a facility-related vulnerability could produce environmental impacts (DOE, 1995a). As discussed above, the canyons would not maintain structural confinement of nuclear materials in a severe earthquake. The estimates of potential environmental consequences from accidents took this acknowledged vulnerability into consideration. If DOE were to design and construct a new facility, there would likely be no environmental consequences from a severe earthquake because a new facility would be designed to withstand such a force.

Similarly, in the Final Interim Management of Nuclear Materials EIS (DOE, 1995a), DOE considered other types of facility vulnerabilities in estimating the potential consequences from accidents. Some examples are (1) a fire that could spread in a facility until it breached containers of nuclear material due to a lack of detection or extinguisher systems, (2) systems that cool nuclear materials stored in tanks that could leak and transfer such material outside the facility before detection, or (3) piping configurations in the canyons that personnel could use inadvertently to transfer solutions of nuclear material to an outside facility tank where they could overflow or spill.

DOE has conducted many reviews to evaluate facility vulnerabilities and has assessed its facilities for compliance with current requirements. DOE has also analyzed the effect on workers and the public from normal and potential accident conditions which could result from operation of facilities with these vulnerabilities. The analysis work was accomplished as a part of ongoing safety review programs and is separate from the NEPA process. Such impact information is represented in the Final Interim Management of Nuclear Materials EIS and in this EIS. The analysis of impacts has, in some cases, prompted DOE to take corrective action based on potential impact alone. For example, DOE has disconnected some tanks of radioactive solutions in the canyons from the canyon cooling system and has isolated canyon tanks by removing interconnected piping to preclude leaks or an inadvertent transfer which could result in a release of radioactive material outside the canyon. In other cases, the potential impact was determined to be small and not sufficient to warrant actions beyond those which could be taken using existing facilities, equipment, and personnel. For example, one vulnerability common to many facilities is that the facility could sustain structural damage in the event of a severe earthquake. This type of earthquake has been estimated to occur once every several thousand years. It would be prohibitively expensive to modify facilities to ensure that no structural damage would occur from such an accident. Rather, DOE has provided mitigation for the consequences of such accidents using engineering safeguards, such as structurally reinforcing tanks, and administrative controls, such as limiting the amount of radioactive material that can be contained in a facility.

H-Canyon Process

The H-Canyon utilized a modified plutonium uranium extraction process (HM process). The HM process unit operations were dissolution, head end, first solvent extraction cycle, second uranium solvent extraction cycle, and second neptunium (or second actinide) solvent extraction cycle. Figure 2-13 shows the historic general H-Canyon process flow.

- Dissolution - Irradiated foreign research reactor spent nuclear fuel was brought into the hot canyon in water-filled casks and through an air lock by railcar. The spent nuclear fuel consists of HEU and LEU aluminum-based irradiated fuel. The spent nuclear fuel was removed from the casks and loaded into a dissolver tank. Heated nitric acid in the tank dissolved the foreign research reactor spent nuclear fuel, resulting in a solution containing enriched uranium, ^{237}Np, small quantities of plutonium, and fission products from the reactor irradiation process, and the cladding material. ^{237}Np should be insignificant in the chemical separation of foreign research reactor spent nuclear fuel.

- Head End - The head end process prepared the target solution for uranium, plutonium, and neptunium separation. First, gelatin was added to precipitate silica and other solid impurities. Then the solution was transferred to a centrifuge, where silica and other impurities were removed as waste, and the clarified product solution was adjusted with nitric acid and water. The wastestream generated from the head-end process was chemically neutralized and sent to high-level waste tanks.

- First Cycle - First cycle operation removed fission products and other chemical impurities, and separated the solution into two product streams for further processing. The chemical properties of acid/solvent/product solutions in contact with each other caused the fission products, the uranium, and the neptunium to separate from the solution containing plutonium.

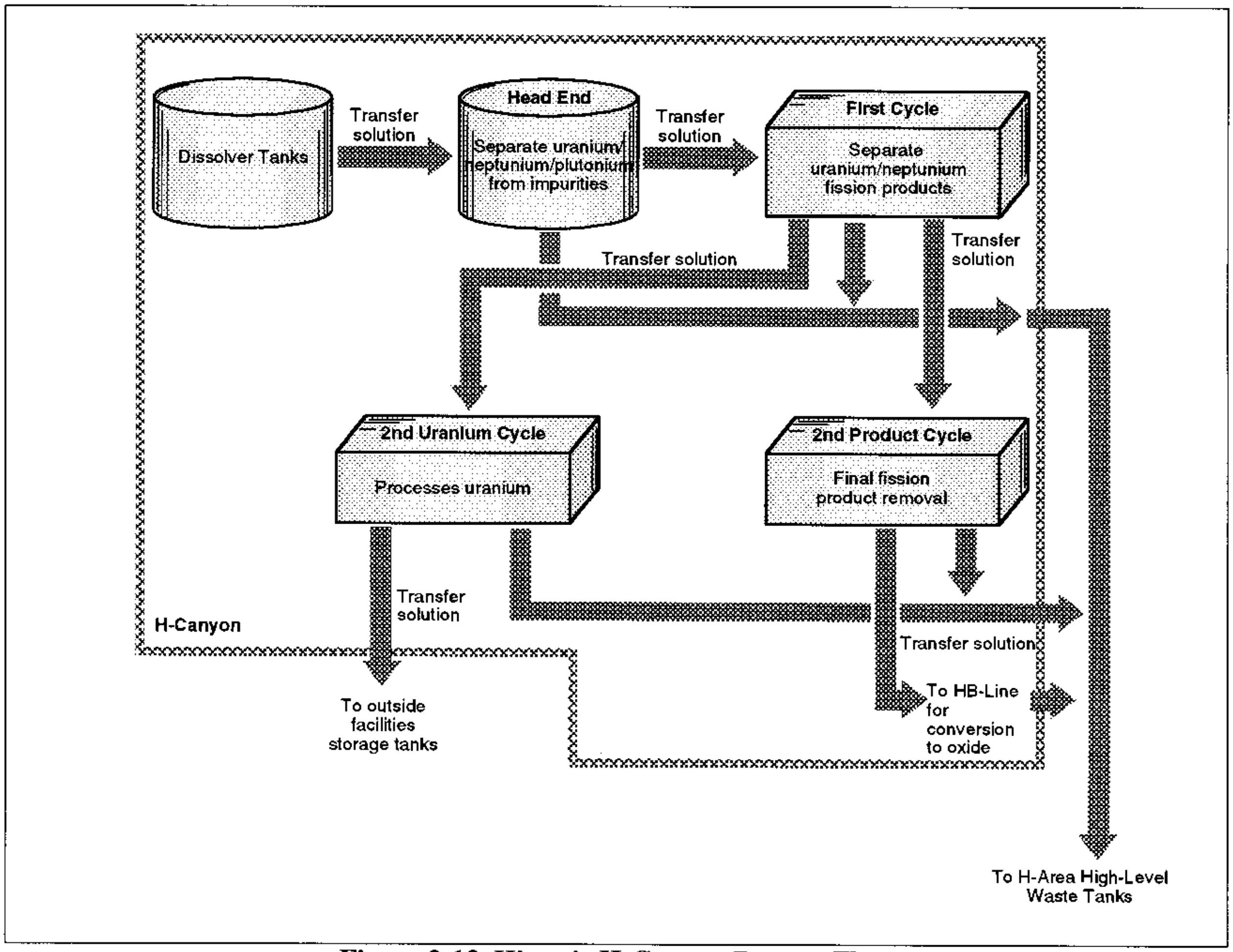


Figure 2-13 Historic H-Canyon Process Flow

- Second Uranium Cycle - The second uranium cycle purified the uranium solution from the first cycle and prepared the uranium for transfer. The purification occurred in a manner similar to that described for the first cycle. The ^{235}U product solution was transferred to storage tanks.

- Second Product Cycle - The second product cycle purified the neptunium solution from the first cycle by removing residual fission products, and prepare the neptunium for transfer. The process occurred in a manner similar to that for the first cycle. The impurities were removed and sent to the low-activity waste unit operation for processing. This cycle would probably be bypassed in the chemical separation of foreign research reactor spent nuclear fuel. Trace neptunium would be discarded as waste.

- High- and Low-Activity Waste - These unit operations reduced the volume of the aqueous streams containing fission products. The streams originate from the separation process unit operations, such as the first cycle. The fission product streams were then separated and sent to high-level waste tanks.

- Solvent Recovery - The primary purpose of this unit operation was to recover and recycle the solvent used in the first cycle. This operation reconditioned and removed impurities from the solvent. The purified solvent was returned to the first and second cycle, cycle reuse and the impurities were transferred to low-activity waste for processing.

F-Canyon Process

The Plutonium Uranium Extraction process at the F-Canyon includes unit operations such as dissolution, head end, first cycle, second uranium cycle, and second plutonium cycle. Unit operations that support the product recovery process were high-activity waste, low-activity waste, and solvent recovery. These were similar to those described for the HM process at H-Canyon with the exception of the Plutonium Uranium Extraction process. In the Plutonium Uranium Extraction process, the second plutonium cycle was equivalent to the second product cycle in the HM process.

Idaho National Engineering Laboratory Facilities

The Idaho Chemical Processing Plant (ICPP) facilities would use a Uranium Extraction process to chemically separate the foreign research reactor spent nuclear fuel for recovery of uranium, and isolation and solidification of the waste fission products resulting from the process. The principal facilities for foreign research reactor spent nuclear fuel chemical separation would be CPP-601, CPP-666, and CPP-602.

Foreign research reactor spent nuclear fuel would be received at the ICPP by truck or rail shipment. Both water-cooled and dry storage facilities would be used. Head-end equipment for initially dissolving or processing the spent nuclear fuel would be available. Aluminum-based clad fuel chemical separation could be conducted in CPP-601. TRIGA-type fuels would be processed in the Fluorinel Dissolution Process in CPP-666. The Fluorinel Dissolution Process cell could require some equipment modifications and additions to accommodate stainless steel-clad dissolution of the TRIGA fuel. However, the process knowledge and equipment is readily available. A new processing facility for uranium fuels is partially completed at the ICPP site. This facility, the Fuel Processing Restoration, is structurally complete but would require completion of services and installation of equipment for foreign research reactor spent nuclear fuel chemical separation.

The high-level liquid waste generated at ICPP during chemical separation of the spent nuclear fuel assemblies would be stored in several large stainless steel underground tanks until it could be processed. Liquid wastes would be converted to a solid calcine form, and then stored dry in bins housed in concrete vaults.

Aluminum and zircaloy fuel processing at ICPP consists of three principal stages. The first is the dissolution stage where fuels were dissolved forming a controlled solution. The second stage is the extraction process, which consists of first, second, and third extraction cycles. These cycles serve to separate and purify the uranium from fission products and material wastes prior to final operations. In the final operation, the solution is fed through a denitrator that conditions the feed material to a solid uranium product that can be packaged, transported, and recycled.

Foreign Reprocessing Facilities

Both France and the United Kingdom have modern fuel cycle facilities and offer reprocessing services to international customers. Either country could sign contracts with foreign research reactor owners/operators for receipt and reprocessing of their spent nuclear fuel, treatment of the waste, and fabrication of fresh fuel. Both France and the United Kingdom would require the country operating the reactor to take back the treated waste.

The French UP1 plant at Marcoule has reprocessed a variety of nuclear fuels, including gas/graphite power reactor fuel and magnesium-clad natural uranium metal fuel. The UP2 plant at La Hague is nearing completion of major renovations that will double its throughput and make it dedicated to oxide fuels. The UP3 plant, also at La Hague, is the newest French reprocessing plant. It started operations in 1990 and is also dedicated to oxide fuels. The French are vigorously engaged in reprocessing commercial power reactor fuel for foreign customers.

The British Prototype Fast Reactor Reprocessing Plant at Dounreay is a small plant associated with the Prototype Fast Reactor. However, it has established a precedent by receiving some research reactor spent nuclear fuel for reprocessing. The Magnox Fuel Reprocessing Plant at Sellafield reprocesses magnesium-clad uranium metal fuel from British gas-cooled reactors. The Thermal Oxide Reprocessing Plant (Thorp) is another large plant at Sellafield for Advanced Gas Reactor and light water reactor fuels. It started operating in January of 1994 and about two-thirds of its scheduled business through 2004 is for foreign customers.

2.6.5.3 Site Management Options

2.6.5.3.1 The Savannah River Site

Only two possible management sites, the Savannah River Site and the Idaho National Engineering Laboratory, would be capable of receiving and managing foreign research reactor spent nuclear fuel at the beginning of the proposed policy implementation period as described in Management Alternative 1.

If the Savannah River Site is the site for managing all DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel would be received and managed there until ultimate disposition. If the Savannah River Site is not the site, foreign research reactor spent nuclear fuel could be received and managed at the Savannah River Site until another site(s) would be ready to receive the foreign research reactor spent nuclear fuel. The construction of new facilities for managing foreign research reactor spent nuclear fuel is estimated to take about 10 years; modifications to existing facilities could take less. For the purposes of the analyses, the period for Phase 1 is assumed to be 10 years. The period following Phase 1 until ultimate disposition is referred to as Phase 2 (approximately 30 years). The amount of spent nuclear fuel that could be received at the Savannah River Site under the basic implementation of Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and discussed in Section 2.6.4.1. Accordingly, the Savannah River Site could receive one-half of the foreign research reactor spent nuclear fuel under the Decentralization and the 1992/1993 Planning Basis alternatives, all of the aluminum-based foreign research reactor spent nuclear fuel under the Regionalization by Fuel Type alternative, only the foreign research reactor spent nuclear fuel from Eastern ports under the Regionalization By Geography alternative, or all foreign research reactor spent nuclear fuel (both aluminum-based and TRIGA) under the Centralization alternative.

As a potential Phase 1 site under Management Alternative 1, the Savannah River Site would receive and manage foreign research reactor spent nuclear fuel at its existing wet storage facilities: RBOF and L-Reactor disassembly basin are considered for this purpose. RBOF is located at the H-Area. It is a facility with provisions for the receipt and storage of irradiated nuclear fuel elements. Since 1963, irradiated spent nuclear fuel elements have been received from offsite reactors and from the Savannah River Site reactors. RBOF provides the capability for underwater unloading of the transportation casks and the handling and storage of the foreign research reactor spent nuclear fuel. The foreign research reactor spent nuclear fuel would be stored in RBOF until the storage capacity is exhausted. Currently, RBOF has space for approximately 1,170 foreign research reactor spent nuclear fuel elements. This capacity could be increased to a total of 2,425 elements by rearranging and consolidating existing inventory. Descriptions of RBOF, the Savannah River Site reactor disassembly basins, and dry cask storage are provided in Appendix F, Section F.3.

The Savannah River Site reactor disassembly basins are not currently configured for storage of MTR type foreign research reactor spent nuclear fuel, however, minor modifications which would provide new storage racks, new handling equipment, safety documentation, etc., along with upgrades in progress to address vulnerabilities associated with water chemistry control, would permit receipt and management of foreign research reactor spent nuclear fuel. Installation of racks equivalent to those in RBOF would provide storage for approximately 20,000 foreign research reactor spent nuclear fuel elements per reactor basin. DOE is considering the L-Reactor disassembly basin for this purpose in this EIS. The modifications to RBOF and L-Reactor disassembly basin are part of the ongoing programs at the site to be performed independent of the proposed action in this EIS.

Between RBOF and the L-Reactor disassembly basin there would be sufficient storage capacity and handling capability to accommodate the receipt and management of foreign research reactor spent nuclear fuel during the estimated 10-year time period for Phase 1.

An additional option to enhance storage capacity during Phase 1 would be to use the existing facilities of RBOF and/or L-Reactor disassembly basin to unload the transportation casks, and provide storage capacity in dry storage casks which would be placed near the existing facility. The storage capacity available and estimated maximum receipt rate of foreign research reactor spent nuclear fuel at the Savannah River Site are shown in Figure F-16 of Appendix F.

As a Phase 2 site, the Savannah River Site would continue to receive foreign research reactor spent nuclear fuel beyond Phase 1 in a new dry storage facility that would be constructed at the H-Area. The location is preferred among a number of sites considered as discussed in Section F.4.1. Foreign research reactor spent nuclear fuel managed during Phase 1 would be transferred to the new facility for management during Phase 2 (approximately 30 years), until ultimate disposition. The dry storage would encompass a number of design examples which were provided in Section 2.6.5.1.1 and Appendix F. Figure 2-14 depicts the facilities and locations considered at the Savannah River Site.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Savannah River Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set which provides a typical, and in many cases, bounding estimate of the resulting impacts.

The specific analysis options under the basic implementation of Management Alternative 1, discussed in Section 2.2.1, are as follows:

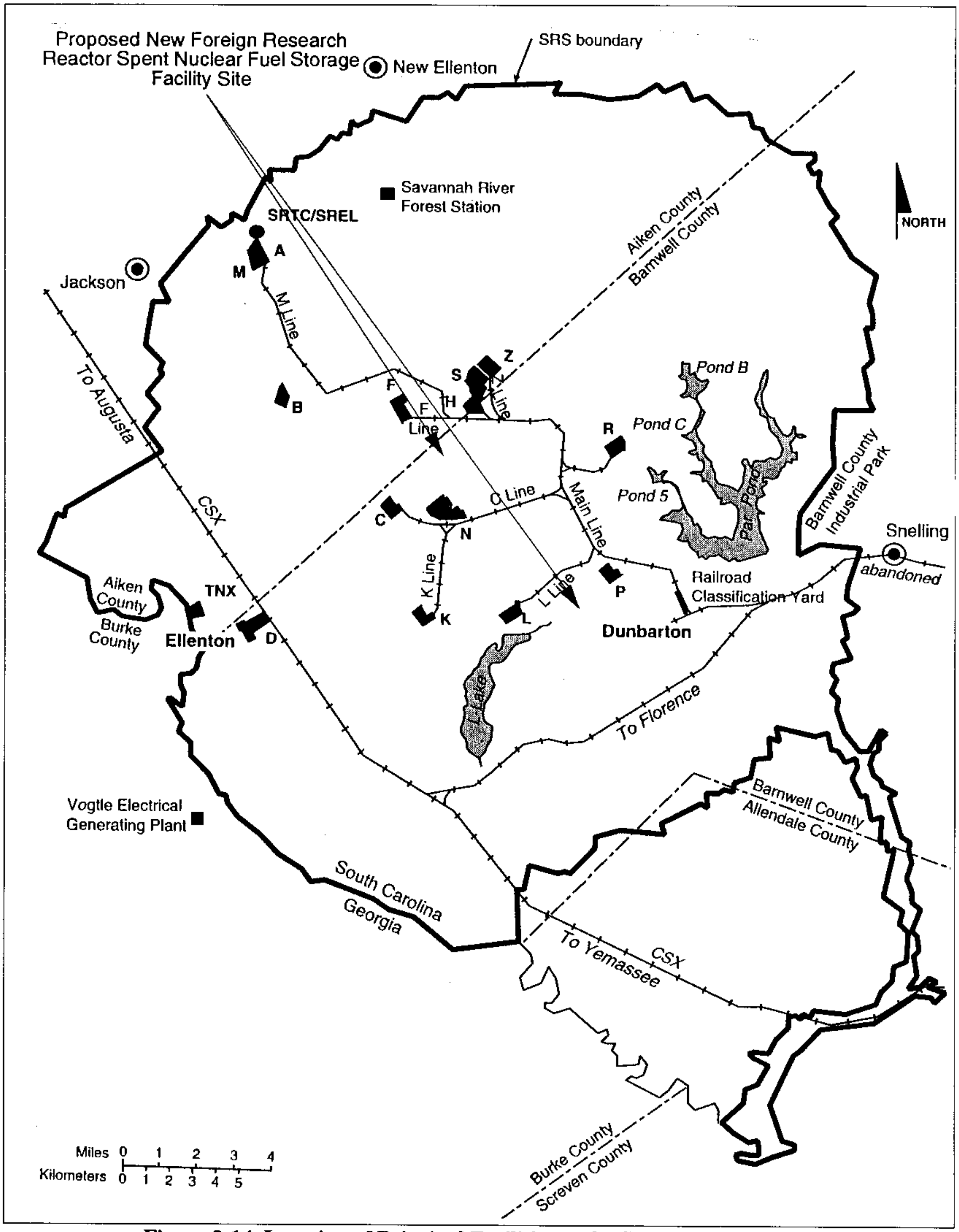


Figure 2-14 Location of Principal Facilities at the Savannah River Site

1A. The Savannah River Site would receive and manage foreign research reactor spent nuclear fuel during Phase 1 and store it at the RBOF and/or the L-Reactor disassembly basin. For the purpose of the analysis, the amount of fuel to be managed is all foreign research reactor spent nuclear fuel that would be received in a 10-year period (17,500 elements). The fuel would be shipped offsite at the end of Phase 1.

1B. Foreign research reactor spent nuclear fuel managed under analysis option 1A would be transferred to a newly constructed dry storage facility, where it would be managed until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes would be received and managed at the new dry storage facility. For the purpose of the analysis, the amount of spent nuclear fuel that would be managed would be all the foreign research reactor spent nuclear fuel (22,700 elements).

The implementation alternatives of Management Alternative 1, discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Savannah River Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Savannah River Site would be likely to receive and manage foreign research reactor spent nuclear fuel in existing facilities during the Phase 1 period. The impacts would be bounded by analysis option 1A (above). Impacts of construction and operation of the dry storage facility considered in analysis option 1B would bound those of the facility required to accommodate this amount of fuel. The spent nuclear fuel would either be shipped offsite after Phase 1, or it would be managed along with the rest of the spent nuclear fuel that would be managed at the Savannah River Site.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Savannah River Site would receive only HEU from the foreign research reactors eligible under the policy. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Savannah River Site would be bounded by analysis options 1A and 1B above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Savannah River Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which, in uranium content, represents approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis options 1A or 1B (above) by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years; and, therefore, the amount of spent nuclear fuel available for management would also be decreased. The impacts from the management of the decreased amount of spent nuclear fuel at the Savannah River Site would be bounded by analysis options 1A or 1B above.

- Under Implementation Subalternative 2b, (Section 2.2.2.2) the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in analysis options 1A or 1B.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be managed in the United States because the foreign research reactors would consider their own alternatives as to whether or not to send the spent nuclear fuel to the United States. The amount of foreign research reactor spent nuclear fuel in this case cannot be quantified. The upper limit, however, is considered under analysis options 1A and 1B (above), which would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management options at the Savannah River Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider wet storage technology for Phase 2 management. DOE would implement this alternative by constructing a new wet storage facility at the H-Area or by using the Barnwell Nuclear Fuels Plant (BNFP), owned by Allied General Nuclear Services. DOE would have to acquire the facility which could be ready for use in approximately 5 years. Therefore, if the Savannah River Site was a selected site under either the Regionalization by Fuel Type or Centralization alternatives, Phase 2 at the Savannah River Site could start as early as 5 years from the start of the implementation period if BNFP were used under this implementation alternative. The new wet storage facility is described in Section 2.6.5.1.2. BNFP is described in Appendix F, Section F.1. For this implementation alternative, an analysis option 1C is considered, which is similar to 1B, as follows:

1C. The spent nuclear fuel managed under analysis option 1A would be transferred to a newly constructed wet storage facility or the BNFP where it would be managed until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would be received and managed at these facilities. For the purpose of the analysis, the amount of spent nuclear fuel that would be managed in these facilities would be all the foreign research reactor spent nuclear fuel (22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. As noted in Section 2.3.6, the Savannah River Site is limited to chemical separation of aluminum-based foreign research reactor spent nuclear fuel.

Under Management Alternative 2, discussed in Section 2.3, DOE and the Department of State would assess the management of foreign research reactor spent nuclear fuel in a foreign location which would include an evaluation of foreign reprocessing with acceptance by the United States of the vitrified high-level waste resulting from reprocessing. The waste would be received and managed at the Defense Waste Processing Facility at the Savannah River Site. DOE and the Department of State estimate that the total volume of the vitrified high-level waste would be about 2.4 m^3 (85 ft^3) and it would fill about 16 European-size canisters. A European-sized canister is about four times smaller than the canister used in the Defense Waste Process Facility at the Savannah River Site. Some modification to the waste handling facility at the Savannah River Site would be necessary to accommodate the smaller canisters.

Under Management Alternative 3 (Hybrid Alternative) discussed in Section 2.4, the Savannah River Site would receive the aluminum-based fuel which would not be reprocessed overseas. This spent nuclear fuel would be processed at the Savannah River Site chemical separation facilities in the same manner as

Implementation Alternative 6 above. The amount of aluminum-based spent nuclear fuel to be chemically separated would be approximately 12,200 elements, 12.9 MTHM, 72 m^3 (2,700 ft^3) as indicated in Table 2-4.

Table 2-11 presents an overview of the foreign research reactor spent nuclear fuel management options, quantities of foreign research reactor spent nuclear fuel assumed for the analysis, and facilities considered.

Table 2-11 Proposed Quantities of Foreign Research Reactor Spent Nuclear Fuel and Management Options at the Savannah River Site

FRR EIS Management Alternative		*FRR SNF Elements*	*Percentage of FRR SNF Total Elements*	*Storage Option/Technology* — *Dry Storage*	*Wet Storage*			*Existing Facilities Plus Dry Cask*[c]	*Chemical Separation*
Management Alternative 1				*New*	*Existing*[a]	*BNFP*[b]	*New*		
All FRR SNF	Phase 1	17,500	77%	NA	A	NA	NA	A	NA
	Phase 2[d]	22,700	100%	A	NA	A	A	NA	NA
Eastern FRR SNF	Phase 1	12,600	56%	NA	A	NA	NA	A	NA
	Phase 2	16,400	72%	A	NA	A	A	NA	NA
Aluminum-based FRR SNF	Phase 1	13,600	60%	NA	A	NA	NA	A	NA
	Phase 2	17,800	78%	A	NA	A	A	NA	NA
Chemical Separation/Storage	Phase 2	17,800	78%	NA	A	NA	NA	A	A
Management Alternative 3									
Aluminum-Based FRR SNF Chemical Separation/Storage		12,300	54%	NA	A	NA	NA	A	A

A = Applicable
NA = Not Applicable
FRR = foreign research reactor
SNF = spent nuclear fuel
[a] *RBOF and L-Reactor basin*
[b] *BNFP could be available for use 5 years after the start of implementation.*
[c] *Dry cask storage would use an existing facility for loading operations.*
[d] *Phase 2 values represent total number of foreign research reactor spent nuclear fuel elements requiring management at the site.*

2.6.5.3.2 Idaho National Engineering Laboratory

Only two possible management sites, the Savannah River Site and the Idaho National Engineering Laboratory, would be capable of receiving and managing foreign research reactor spent nuclear fuel at the beginning of the proposed policy implementation period.

If the Idaho National Engineering Laboratory is the site for managing all DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel would be received and managed there until ultimate disposition. If the Idaho National Engineering Laboratory is not the site, foreign research reactor spent nuclear fuel could be received and managed at the Idaho National Engineering Laboratory until another site(s) would be ready to receive the foreign research reactor spent nuclear fuel. The construction of new facilities for managing foreign research reactor spent nuclear fuel is estimated to take about 10 years; this period is referred to as Phase 1. The period following Phase 1 until ultimate disposition is referred to as Phase 2 (approximately 30 years). The amount of spent nuclear fuel that could be received at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). Accordingly, the Idaho National Engineering Laboratory could receive one-half of the foreign research reactor spent nuclear fuel under the Decentralization and the 1992/1993 Planning Basis alternatives, all of the TRIGA-type foreign research reactor spent nuclear fuel under the Regionalization by Fuel Type alternative, only the foreign research reactor spent nuclear fuel from Western ports under the Regionalization By Geography alternative, or all foreign research reactor spent nuclear fuel under the Centralization alternative.

As a potential Phase 1 site, the Idaho National Engineering Laboratory would receive and manage foreign research reactor spent nuclear fuel at existing dry and wet storage facilities. The existing facilities identified for this purpose would be the Fluorinel Dissolution and Fuel Storage (FAST) facility in CPP-666, the Irradiated Fuel Storage Facility (IFSF) in CPP-603, and the CPP-749 storage area. Descriptions of these facilities are provided in Appendix F, Section F.3.

The FAST facility is a modern underwater storage facility which has been used in the past for receipt and storage of foreign research reactor spent nuclear fuel. It has the capability to receive and unload spent nuclear fuel casks at a rate of approximately five per week. Storage capacity for up to 8,400 foreign research reactor spent nuclear fuel elements could be provided in a 10-year period by using the spent nuclear fuel storage racks that would be installed. The capability of the FAST facility to receive foreign research reactor spent nuclear fuel in the near term is limited due to the number of activities scheduled through FY 1998. Considering these activities, DOE estimates that 3,600 elements could be received by the end of 1999 at the FAST facility.

The IFSF is a shielded dry storage vault originally constructed for Fort St. Vrain reactor fuel. The storage capacity available is for approximately 9,000 foreign research reactor spent nuclear fuel elements. However, as with the FAST facility, many activities are already scheduled for the facility. Considering these activities, foreign research reactor spent nuclear fuel could not be received until sometime in FY 1997 and could continue at the rate of 50 shipments per year (approximately 1,500 elements) thereafter.

The CPP-749 underground spent nuclear fuel storage area is a dry storage facility with a remote unloading area and vault storage. With some refurbishment it could provide space for 3,600 elements starting in FY 1998. The spent nuclear fuel would go through the IFSF to be placed in baskets and transferred to a compatible storage cask. The refurbishments of existing facilities are part of the ongoing programs at the site, to be performed independent of the proposed action in this EIS.

Between these facilities there is sufficient storage space and handling capacity to accommodate the receipt and management of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory during the Phase 1 period. The storage capacity available and estimated maximum receipt rate of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory are shown in Figure F-18 of Appendix F.

An additional option to enhance storage capacity during Phase 1 would be to use the existing facilities to unload the transportation casks, and provide storage capacity in dry storage casks which would be placed near the existing facility. Descriptions of the Idaho National Engineering Laboratory existing facilities are provided in Appendix F. The location of these facilities at the Idaho National Engineering Laboratory are shown in Figure 2-15.

As a Phase 2 site, the Idaho National Engineering Laboratory would continue to receive and manage foreign research reactor spent nuclear fuel at existing facilities until a new dry storage facility were to become operational at the site. Foreign research reactor spent nuclear fuel managed at existing facilities

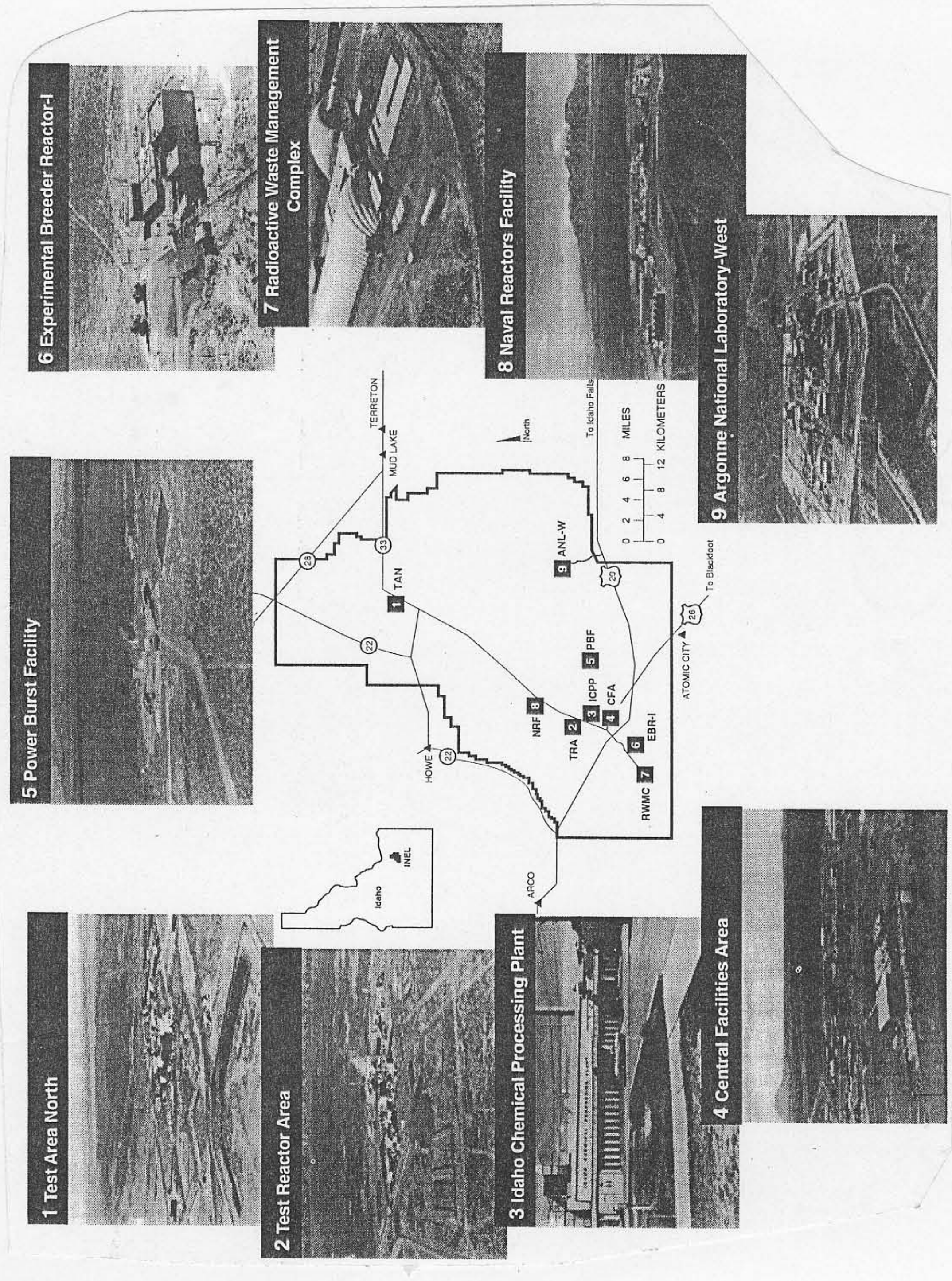


Figure 2-15 Location of Principal Facilities at the Idaho National Engineering Laboratory

would then be transferred to the new facility where it would remain until ultimate disposition. The new facility would also receive foreign research reactor spent nuclear fuel shipments directly from ports after the 10-year policy period. Dry storage encompasses both the dry vault design and the dry cask design as described in Section 2.6.5.1.1.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set which provides a typical, and in many cases, bounding estimate of the resulting impacts.

The specific analysis options under the basic implementation of Management Alternative 1, discussed in Section 2.2.1, are as follows:

2A. The Idaho National Engineering Laboratory would receive and manage foreign research reactor spent nuclear fuel during Phase 1 and store it at the FAST, the IFSF, and/or the CPP-749 facilities. For the purpose of the analysis, the amount of fuel to be managed is all foreign research reactor spent nuclear fuel that would be received in a 10-year period (17,500 elements). The fuel would be shipped offsite at the end of Phase 1.

2B. Foreign research reactor spent nuclear fuel managed under analysis option 2A would be transferred to a newly constructed dry storage facility where it would be managed until ultimate disposition. Spent nuclear fuel arriving at the United States after Phase 1 concludes would be received and managed at the new dry storage facility until ultimate disposition. For the purpose of the analysis, the amount of spent nuclear fuel that would be stored in the dry storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements).

The implementation alternatives of Management Alternative 1 discussed in Section 2.2.2 introduce additional analysis options that could be considered for the Idaho National Engineering Laboratory as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Idaho National Engineering Laboratory would be likely to receive and manage foreign research reactor spent nuclear fuel in existing facilities during the Phase 1 period. The impacts would be bounded by analysis option 2A above. The dry storage facility considered in analysis option 2B would be sized to accommodate this amount of fuel. The fuel would either be shipped offsite after Phase 1 or it would be managed along with the rest of the spent nuclear fuel that would be managed at the Idaho National Engineering Laboratory.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Idaho National Engineering Laboratory would receive only HEU from the reactors eligible under the proposed action. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Idaho National Engineering Laboratory would be bounded by analysis options 2A and 2B above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Idaho National Engineering Laboratory would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents, in uranium content, approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis options 2A or 2B (above) by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years and therefore the amount of spent nuclear fuel available for management would also be decreased. The impacts from the management of the decreased amount of spent nuclear fuel at the Idaho National Engineering Laboratory would be bounded by analysis options 2A or 2B above.

- Under Implementation Subalternative 2b, (Section 2.2.2.2) the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in analysis options 2A or 2B.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be managed in the United States because the foreign research reactors would consider their own alternatives about whether to send the spent nuclear fuel to the United States. The amount of foreign research reactor spent nuclear fuel in this case cannot be quantified. The upper limit, however, is considered under analysis options 2A or 2B which would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management options at the Idaho National Engineering Laboratory.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Idaho National Engineering Laboratory for Phase 2 until ultimate disposition. The new wet storage facility is described in Section 2.6.5.1.2. For this implementation alternative, an analysis option 2C, which is similar to analysis option 2B, is considered as follows:

2C. The spent nuclear fuel managed under analysis option 2A would be transferred to a newly constructed wet storage facility where it would be managed until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would be received and managed at the new wet storage facility until ultimate disposition. For the purpose of the analysis, the amount of spent nuclear fuel that would be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. As noted in the discussion in Section 2.3.6, chemical separation of both aluminum-based and TRIGA foreign research reactor spent nuclear fuel is evaluated for the Idaho National Engineering Laboratory.

Under Management Alternative 3 (Hybrid Alternative), discussed in Section 2.4, the Idaho National Engineering Laboratory would receive the foreign research reactor TRIGA spent nuclear fuel. This spent nuclear fuel would be managed at the Idaho National Engineering Laboratory in existing facilities until final disposition. The amount of TRIGA spent nuclear fuel to be stored would be 4,900 elements, 1.0 MTHM, and about 4 m^3 (200 ft^3) as indicated in Table 2-4.

Table 2-12 presents an overview of the foreign research reactor spent nuclear fuel management options, quantities of foreign research reactor spent nuclear fuel assumed for the analysis, and facilities considered.

Table 2-12 Proposed Quantities of Foreign Research Reactor Spent Nuclear Fuel and Management Options at the Idaho National Engineering Laboratory

FRR EIS Management Alternative		*FRR SNF Elements*	*Percentage of Total FRR SNF Elements*	*Storage Option/Technology*					
				Dry Storage		*Wet Storage*		*Existing Facilities plus Casks*[c]	*Chemical Separation*
Management Alternative 1				*New*	*Existing*[a]	*New*	*Existing*[b]		
All FRR SNF	Phase 1	17,500	77%	NA	A	NA	A	A	NA
	Phase 2[d]	22,700	100%	A	NA	A	NA	NA	NA
Western FRR SNF	Phase 1	4,800	21%	NA	A	NA	A	A	NA
	Phase 2	6,300	28%	A	NA	A	NA	NA	NA
TRIGA FRR SNF	Phase 1	3,800	17%	NA	NA	NA	A	A	NA
	Phase 2	4,900	22%	NA	A	A	NA	NA	NA
Chemical Separation/Storage		22,700	100%	NA	A	NA	A	A	A
Management Alternative 3									
TRIGA FRR SNF/ Storage		4,900	22%	NA	A	NA	A	A	NA

A = Applicable
NA = Not Applicable
FRR = foreign research reactor
SNF = spent nuclear fuel
[a] *IFSF, CPP-749*
[b] *FAST*
[c] *Existing facilities augmented by dry cask storage.*
[d] *Phase 2 values represent total number of foreign research reactor spent nuclear fuel elements requiring management at the site.*

2.6.5.3.3 The Hanford Site

Options for receiving and managing foreign research reactor spent nuclear fuel at the Hanford Site are primarily dictated by the Programmatic SNF&INEL Final EIS (DOE, 1995c) alternatives, and the lack of suitable facilities at the Hanford Site to receive foreign research reactor spent nuclear fuel at the beginning of the proposed policy period.

If the Hanford Site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period (Phase 1) required for the Hanford Site to have new facilities constructed and operational to accommodate the spent nuclear fuel. As discussed in previous sections, Phase 1 is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2) the Hanford Site would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National

Engineering Laboratory and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Hanford Site until ultimate disposition.

The amount of spent nuclear fuel that would be received and managed at the Hanford Site under the basic implementation of Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). Accordingly, in Phase 2, the Hanford Site could receive the TRIGA foreign research reactor spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1, Western foreign research reactor spent nuclear fuel under the Regionalization By Geography alternative, or all foreign research reactor spent nuclear fuel under the Centralization alternative.

Under the basic implementation of Management Alternative 1, and as a Phase 2 site, the Hanford Site would receive and manage foreign research reactor spent nuclear fuel at a new dry storage facility constructed at the 200 Area Plateau or the Fuel Material Examination Facility (FMEF), which is a partially completed, large, hot cell facility. The new dry storage facility is described in Section 2.6.5.1.1. Description of the FMEF is provided in Appendix F, Section F.3. Figure 2-16 displays the potential construction locations for foreign research reactor spent nuclear fuel storage at the Hanford Site. FMEF is located near the Fast Flux Test Facility.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Hanford Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis option under the basic implementation of Management Alternative 1, discussed in Section 2.2.1, is as follows:

3A. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Hanford Site where it would be managed at a new dry storage facility constructed either at the 200 Area Plateau or at FMEF. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new facility until ultimate disposition. For the purposes of the analysis, the total amount of spent nuclear fuel that would be managed in the dry storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements). If the Hanford Site were to receive TRIGA from the Idaho National Engineering Laboratory or only western spent nuclear fuel, the dry storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

The implementation alternatives of Management Alternative 1, which are discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Hanford Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Hanford Site would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site and manage it in facilities sized for this amount. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 3A.

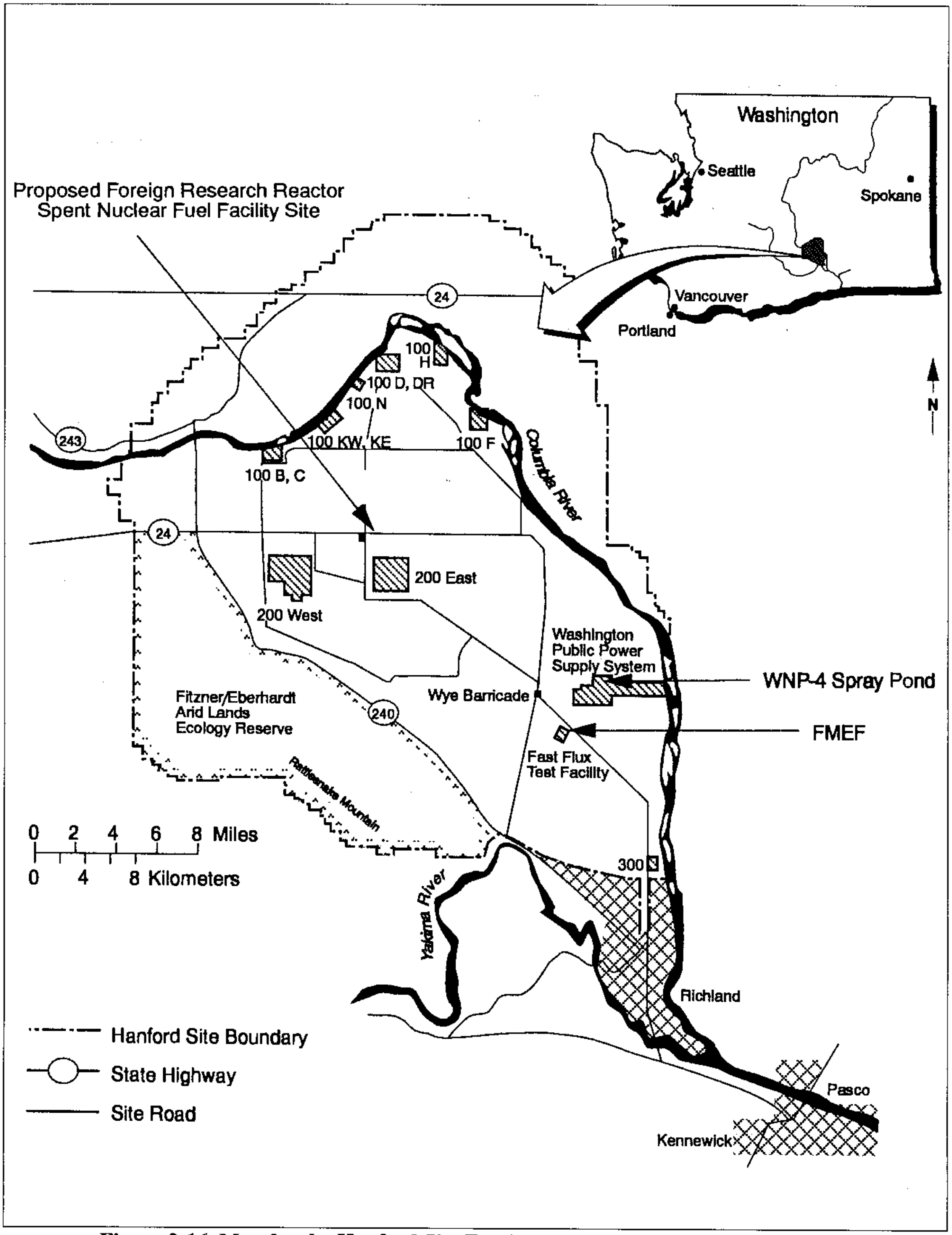


Figure 2-16 Map for the Hanford Site Foreign Research Reactor Spent Nuclear Fuel Storage (in the 200 Areas)

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Hanford Site would receive only HEU from the Idaho National Engineering Laboratory and/or the Savannah River Site. The amount would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Hanford Site would be bounded by analysis option 3A above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Hanford Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 3A by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years; and, therefore, the amount of spent nuclear fuel available for management would also be decreased. In this case, the Hanford Site would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Hanford Site would be bounded by analysis option 3A above.

- Under Implementation Subalternative 2b (Section 2.2.2.2), the acceptance of a small portion of the fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in option 3A.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be managed in the United States as the foreign research reactors would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of spent nuclear fuel in this case cannot be quantified; however, the upper limit, which is considered under analysis option 3A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management options at the Hanford Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Hanford Site for Phase 2 until ultimate disposition. For this implementation alternative, an analysis option 3B, which is similar to 3A, is considered as follows:

3B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 is shipped to the Hanford Site where it would be managed at a new wet storage facility constructed at either the 200 Area Plateau or the WNP-4 Spray Pond facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel to be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements). If the Hanford Site were to receive only TRIGA fuel from the Idaho National Engineering Laboratory, or only western fuel, the wet storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Hanford Site would not be considered as a site for chemical separation. The Hanford Site is also not considered under the Hybrid Alternative discussed in Section 2.4.

Table 2-13 presents an overview of the foreign research reactor spent nuclear fuel management options, quantities of foreign research reactor spent nuclear fuel assumed for the analysis, and facilities considered.

Table 2-13 Proposed Quantities of Foreign Research Reactor Spent Nuclear Fuel and Management Options at Hanford Site

Management Alternative 1		*FRR SNF Element*	*Percentage of Total FRR SNF Elements*	*Storage Option/Technology*			
				Dry Storage		*Wet Storage*	
				New at FMEF	*New*	*New at WNP-4*	*New*
All FRR SNF	Phase 2	22,700	100%	A	A	A	A
Western FRR SNF	Phase 2	6,300	28%	A	A	A	A
TRIGA FRR SNF	Phase 2	4,900	22%	A	A	A	A

A = Applicable
FRR = foreign research reactor
SNF = spent nuclear fuel

2.6.5.3.4 The Oak Ridge Reservation

The options for receiving and managing foreign research reactor spent nuclear fuel at the Oak Ridge Reservation are primarily dictated by the Programmatic SNF&INEL Final EIS (DOE, 1995c) alternatives and the lack of suitable facilities at the Oak Ridge Reservation to receive foreign research reactor spent nuclear fuel at the beginning of the proposed policy period.

If the Oak Ridge Reservation is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period (Phase 1) required for the Oak Ridge Reservation to construct and to place in operation new facilities to accommodate the spent nuclear fuel. As discussed in previous sections, Phase 1 is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2) the Oak Ridge Reservation would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Oak Ridge Reservation until ultimate disposition.

The amount of spent nuclear fuel that would be received and managed at the Oak Ridge Reservation under the basic implementation of Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). Accordingly, in Phase 2, the Oak Ridge Reservation could receive aluminum-based foreign research reactor spent nuclear fuel managed at the Savannah River

Site during Phase 1, Eastern foreign research reactor spent nuclear fuel under the Regionalization by Geography alternative, or all foreign research reactor spent nuclear fuel under the Centralization alternative.

Under the basic implementation of Management Alternative 1, and as a Phase 2 site, the Oak Ridge Reservation would receive and manage foreign research reactor spent nuclear fuel at a new dry storage facility to be constructed at the West Bear Creek Valley Site. The location is preferred among the four locations considered in a siting study performed for spent nuclear fuel management (MMES, 1994). The four locations considered are shown in Figure 2-17. Description of the new dry storage facility is provided in Section 2.6.5.1.1.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation is based on the above considerations. The analysis options selected do not represent all possible combinations but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis option under the basic implementation of Management Alternative 1, discussed in Section 2.2.1, is as follows:

4A. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Oak Ridge Reservation where it would be managed at a new dry storage facility until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new facility until ultimate disposition. For the purposes of the analysis, the total amount of spent nuclear fuel that would be managed in the dry storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements).

The implementation alternatives of Management Alternative 1, which are discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Oak Ridge Reservation as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Oak Ridge Reservation would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and manage it in facilities sized for this amount. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 4A.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Oak Ridge Reservation would receive only HEU from the Idaho National Engineering Laboratory and/or the Savannah River Site. The amount of spent nuclear fuel would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Oak Ridge Reservation would be bounded by analysis option 4A above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Oak Ridge Reservation would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The analysis assumes that the receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 4A by a small percentage.

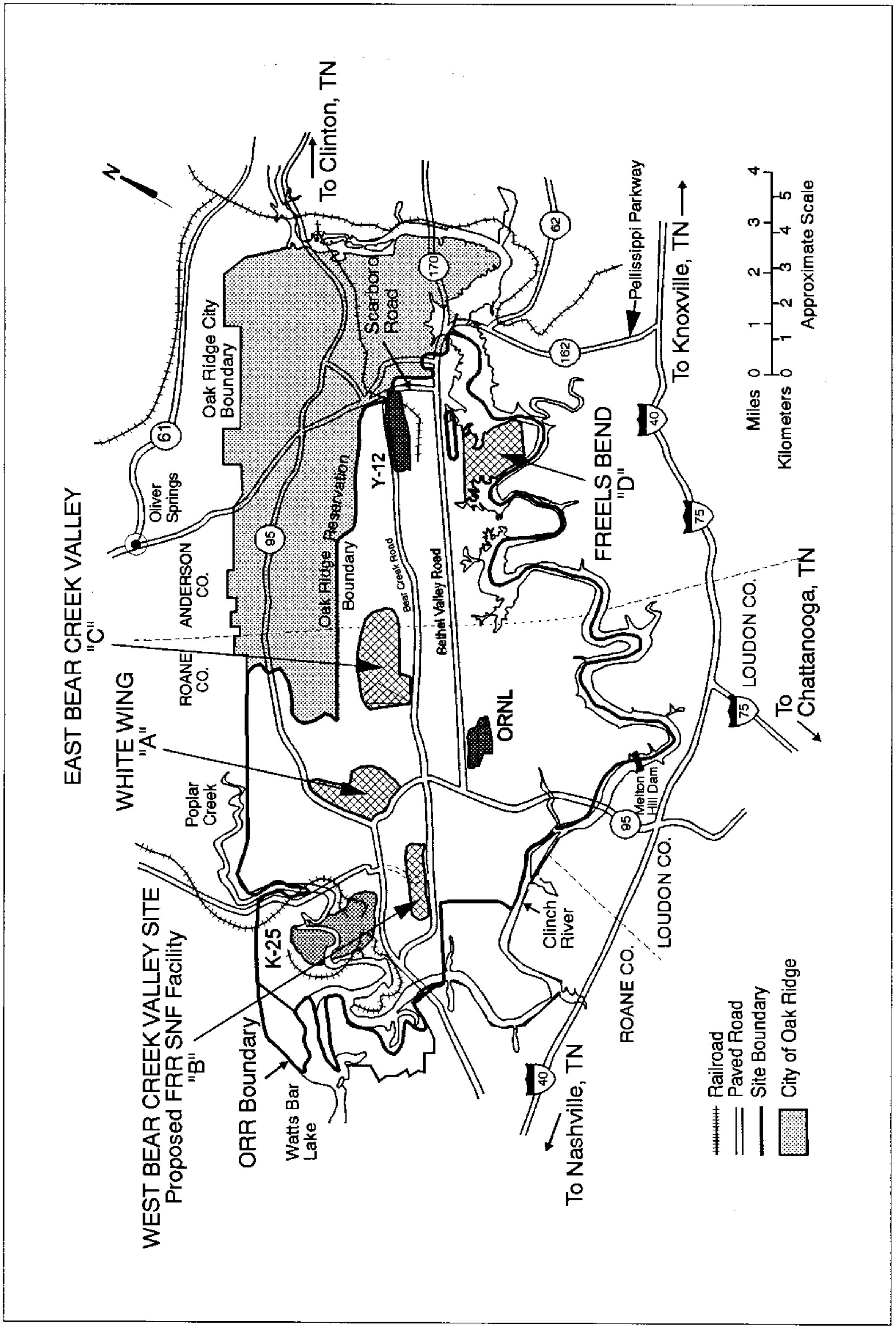


Figure 2-17 Candidate Sites at the Oak Ridge Reservation for Foreign Research Reactor Spent Nuclear Fuel Storage

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years; and, therefore, the amount of spent nuclear fuel available for management would also be decreased. In this case, the Oak Ridge Reservation would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Oak Ridge Reservation would be bounded by analysis option 4A above.

- Under Implementation Subalternative 2b (Section 2.2.2.2), the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in option 4A.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be managed in the United States as the foreign research reactors would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of spent nuclear fuel in this case cannot be quantified; however, the upper limit, which is considered under analysis option 4A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management options at the Oak Ridge Reservation.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Oak Ridge Reservation for Phase 2 until ultimate disposition. For this implementation alternative an analysis option 4B, which is similar to 4A, is considered as follows:

4B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Oak Ridge Reservation where it would be managed at a new wet storage facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel to be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Oak Ridge Reservation would not be considered as a site for chemical separation. The Oak Ridge Reservation is also not considered under the Hybrid Alternative discussed in Section 2.4.

Table 2-14 presents an overview of the foreign research reactor spent nuclear fuel management options, quantities of foreign research reactor spent nuclear fuel assumed for the analysis, and facilities considered.

Table 2-14 Proposed Quantities of Foreign Research Reactor Spent Nuclear Fuel and Management Options at Oak Ridge Reservation

Management Alternative 1		*FRR SNF Elements*	*Percentage of Total FRR SNF Elements*	*Storage Option/Technology*	
				Dry Storage (New)	*Wet Storage (New)*
All FRR SNF	Phase 2[a]	22,700	100%	A	A
Eastern FRR SNF	Phase 2	16,400	72%	A	A
Aluminum-based FRR SNF	Phase 2	17,800	78%	A	A

A = Applicable
FRR = foreign research reactor
SNF = spent nuclear fuel
[a] *Phase 2 values represent total number of foreign research reactor spent nuclear fuel elements requiring management at the site.*

2.6.5.3.5 The Nevada Test Site

The options for receiving and managing foreign research reactor spent nuclear fuel at the Nevada Test Site are primarily dictated by the Programmatic SNF&INEL Final EIS (DOE, 1995c) alternatives, and the lack of suitable facilities at the Nevada Test Site to receive foreign research reactor spent nuclear fuel at the beginning of the proposed policy period.

If the Nevada Test Site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period (Phase 1) required for the Nevada Test Site to have new facilities constructed and operational to accommodate the spent nuclear fuel. As discussed in previous sections, Phase 1 is estimated to be about 10 years. At the end of Phase 1 (i.e., start of Phase 2) the Nevada Test Site would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Nevada Test Site until ultimate disposition.

Although the Nevada Test Site has no existing facilities to receive foreign research reactor spent nuclear fuel at the beginning of the management period, it has facilities that could be modified to receive foreign research reactor spent nuclear fuel within 5 years. These facilities are large hot cells located in the Nevada Research and Development Area on Jackass Flats. Presently these facilities (e.g., the Engine Maintenance and Disassembly [E-MAD] facility) have little usage, but some are in acceptable condition. To use the E-MAD facility, a small pool would have to be constructed to be used for transferring the spent nuclear fuel from the transportation casks to containers designed for dry storage. A description of the E-MAD facility is included in Appendix F (Section F.1). The E-MAD facility could be ready within 5 years of the start of the proposed policy period.

The amount of spent nuclear fuel that would be received and managed at the Nevada Test Site under the basic implementation of Management Alternative 1, is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). Accordingly, during Phase 2, the Nevada Test Site could receive TRIGA foreign research reactor spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1, only Western foreign research reactor spent nuclear fuel under the Regionalization By Geography alternative, or all foreign research reactor spent nuclear fuel under the Centralization alternative.

As a Phase 2 site, the Nevada Test Site would receive and manage foreign research reactor spent nuclear fuel at a newly constructed dry storage facility or a modified E-MAD facility. Description of the new dry storage facility is provided in Section 2.6.5.1.1. Figure 2-18 displays the potential construction location and the area where the E-MAD facility is located at the Nevada Test Site.

The analysis of potential environmental impacts from management of foreign research reactor spent nuclear fuel at the Nevada Test Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis option under the basic implementation of Management Alternative 1, discussed in Section 2.2.1, is as follows:

5A. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Nevada Test Site where it would be managed at a new dry storage facility or a modified E-MAD facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new or E-MAD facility until ultimate disposition. For the purposes of the analysis, the total amount of spent nuclear fuel that would be stored would be all the foreign research reactor spent nuclear fuel (22,700 elements).

The implementation alternatives of Management Alternative 1, discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Nevada Test Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Nevada Test Site would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and manage it in facilities sized for the reduced amount. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 5A.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Nevada Test Site would receive from the Idaho National Engineering Laboratory and/or the Savannah River Site only HEU. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel would be bounded by analysis option 5A above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Nevada Test Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 5A by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years; and, therefore, the amount of spent nuclear fuel available for management would also be decreased. In such a case, the Nevada Test Site would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the

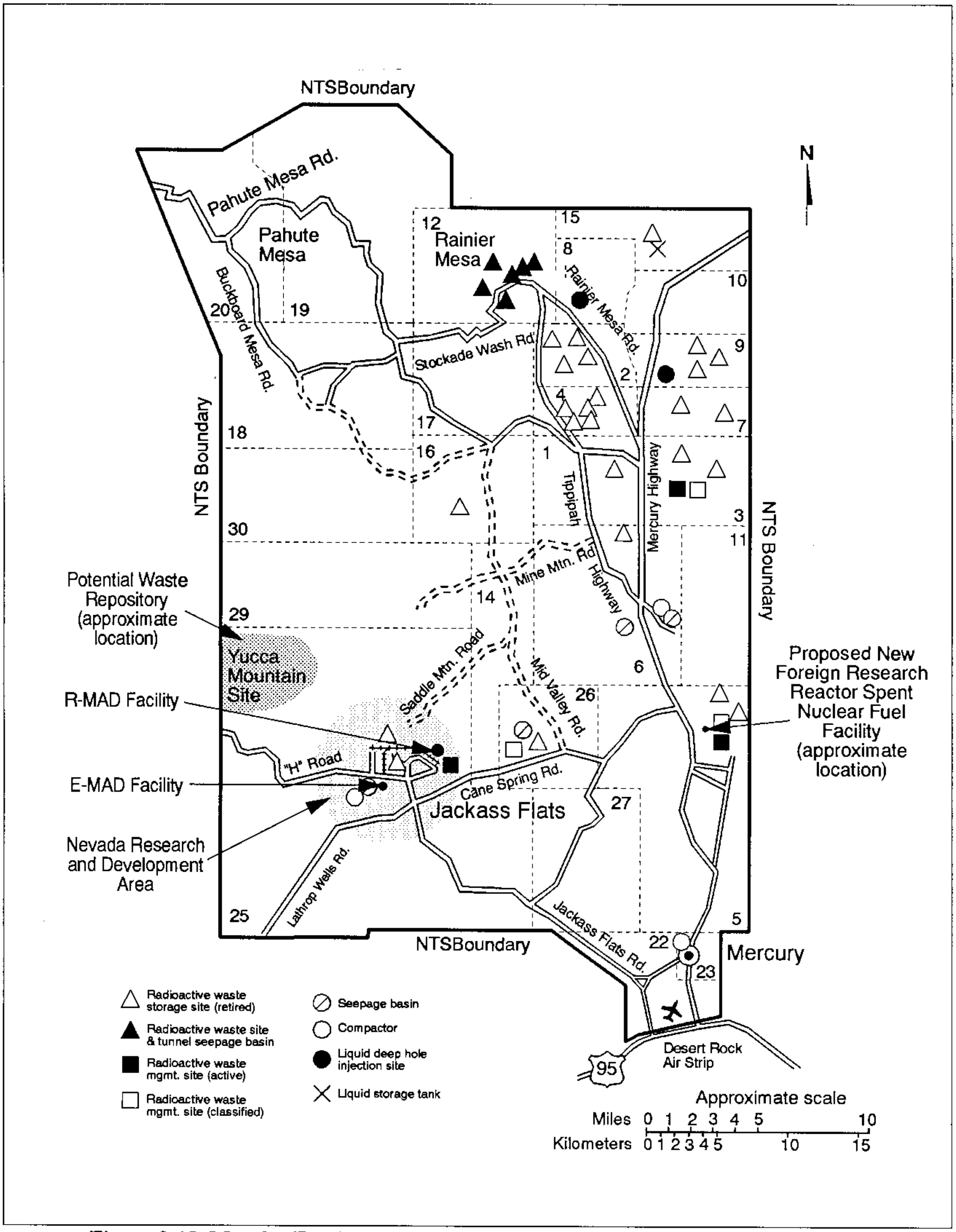


Figure 2-18 Map for Foreign Research Reactor Spent Nuclear Fuel Storage at the Nevada Test Site

Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Nevada Test Site would be bounded by analysis option 5A above.

- Under Implementation Subalternative 2b (Section 2.2.2.2), the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in analysis option 5A.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. The various arrangements would affect the amount of spent nuclear fuel that would be managed in the United States as the foreign research reactors would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of spent nuclear fuel in this case cannot be quantified; however, the upper limit, considered under analysis option 5A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management options at the Nevada Test Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Nevada Test Site for Phase 2 until ultimate disposition. For this implementation alternative an analysis option 5B, which is similar to 5A, is considered as follows:

5B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Nevada Test Site where it would be managed at a new wet storage facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel that would be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements). If the Nevada Test Site were to receive only TRIGA spent nuclear fuel from the Idaho National Engineering Laboratory or only western spent nuclear fuel, the wet storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Nevada Test Site would not be considered as a site for chemical separation. The Nevada Test Site is also not considered for the Hybrid Alternative discussed in Section 2.4.

Table 2-15 presents an overview of the foreign research reactor spent nuclear fuel management options, quantities of foreign research reactor spent nuclear fuel assumed for the analysis, and facilities considered.

Table 2-15 Proposed Quantities of Foreign Research Reactor Spent Nuclear Fuel and Management Options at the Nevada Test Site

Management Alternative 1		*FRR SNF Elements*	*Percentage of Total FRR SNF Elements*	*Storage Option/Technology*		
				Dry Storage		*Wet Storage*
				E-MAD[a]	*New*	*New*
All FRR SNF	Phase 2[b]	22,700	100%	A	A	A
Western FRR SNF	Phase 2	6,300	28%	A	A	A
TRIGA FRR SNF	Phase 2	4,900	22%	A	A	A

A = Applicable
NA = Not Applicable
FRR = foreign research reactor
SNF = spent nuclear fuel

[a] *E-MAD could be available for use five years after the start of implementation.*

[b] *Phase 2 values represent total number of foreign research reactor spent nuclear fuel elements requiring management at the site.*

2.7 Characteristics of Emergency Management and Response

This section addresses the emergency management and response infrastructure that exists to support the possible implementation of those management alternatives of the proposed action that would have an impact in the United States. This section considers emergency management and response at the ports of entry, along ground transport routes, and at the management sites.

2.7.1 DOE and the National Response System

In the United States, State and local governments are responsible for emergency management and response programs. These programs must be capable of managing all hazards ranging from natural disasters to hazardous material incidents on a day-to-day basis. In order to maintain these programs, various State, Tribal, and local governments receive Federal funding. DOE historically has provided a variety of support to governmental jurisdictions in fulfilling its responsibilities under regulatory and National emergency plan taskings (FEMA, 1994; Rogoff, 1994; and DOE, 1994g).

There are three national emergency response plans (i.e., Federal Response Plan, Federal Radiological Emergency Response Plan, and the National Contingency Plan) under which DOE provides radiological monitoring and assessment assistance. Under these plans, DOE provides technical advice and assistance to State, Tribal, and local agencies involved with a radiological incident (DOE, 1989). For a foreign research reactor spent nuclear fuel incident, DOE actions would be guided by the Federal Radiological Emergency Response Plan (FEMA, 1985) and its own internal emergency management and response system.

DOE maintains an emergency management and response system that is based on regulatory requirements as outlined in various DOE Orders (e.g., DOE Order Series 5500 and 5530). These orders require an emergency management and response system that generally follows the models and practices established by the Federal Emergency Management Agency, National Fire Protection Association, American National Standards Institute, and National Council on Radiation Protection and Measurements.

2.7.2 Foreign Research Reactor Spent Nuclear Fuel Transportation

Foreign research reactor operators, their shipping agents, and commercial carriers would have the primary responsibility to coordinate and arrange all activities associated with foreign research reactor spent nuclear fuel shipments and cask return including emergency management and response. DOE, along with other Federal agencies (e.g., Department of Transportation, NRC, Federal Emergency Management Agency, U.S. Department of Defense, and the U.S. Environmental Protection Agency) would provide support and assistance to State, Tribal, and local government agencies responsible for responding to a foreign research reactor spent nuclear fuel incident.

DOE fulfills its role and responsibilities as the Federal agency tasked with developing and maintaining a capability to safely manage spent nuclear fuel (DOE, 1995c), in part by setting overall spent nuclear fuel program management responsibility and policy for transportation and emergency management and response; resolving policy questions; issuing guidance; providing information; and accomplishing oversight by including regulatory compliance requirements in its spent nuclear fuel related contracts and by monitoring the performance of those involved.

According to DOE records, from 1985 to 1993 there were 102,213 DOE shipments consisting of 1,009,357.6 metric tons (1,112,626 tons) of radioactive material. Of these, 457 shipments, containing 13,176.86 metric tons (14,525 tons) were spent nuclear fuel (these weights include the packaging) (DOE, 1994d). To date, there are no records of radiological fatalities that have occurred in the United States due to transportation accidents. To date, no spent nuclear fuel transportation cask has ever been punctured or released any of its radioactive contents, even in actual highway accidents (NRC, 1993).

2.7.3 External Coordination

Historically, DOE ensures coordination with various organizations and agencies through its interaction with Government, national, and local groups such as the Southern States Energy Board and Western Governors' Association, among others.

The primary responsibility for developing and maintaining a radiological hazardous materials emergency response capability is vested in State, Tribal, and local agencies. DOE, on an "as needed," case-by-case basis, has helped State, Tribal, and local agencies prepare for response to potential accidents involving DOE radioactive material shipments (including spent nuclear fuel). As with the Urgent Relief foreign research reactor spent nuclear fuel transportation effort, DOE has offered various types of technical assistance to the affected jurisdictions (SSEB, 1994).

One example of this partnership with State and local governments occurred in August 1994. In support of Urgent Relief safe transportation, special Radiological Emergency Training for Local Responders and Emergency Response Workshop courses were conducted by DOE for approximately 160 local responders from North Carolina and South Carolina (Analysas Corporation, 1994).

2.7.3.1 Financial and Technical Assistance to States and Tribes

DOE provides funding to States and Tribes through the Office of Environmental Management and the Office of Civilian Radioactive Waste Management to assist with transportation related issues. While some of these funding efforts are not directly related to spent nuclear fuel shipments, they do enhance a jurisdiction's emergency management and response capabilities (DOE, 1994g). Financial assistance to States and Tribes for Transportation programs for fiscal year 1994 is shown in Table 2-16 (DOE, 1994g).

Table 2-16 DOE Summary of Financial Assistance to States and Indian Tribes

Total Allocations for Transportation Programs: FY 1994	
Activity	*Amount*
Waste Isolation Pilot Plant	$1,410,848
Cesium Shipment Support	330,000
Spent Fuel Shipment Support	125,000
Transportation External Coordination Working Group[a]	34,921
Urban Energy & Transportation Corporation[b]	150,137
Office of Civilian Radioactive Waste Management	1,332,000
State of Washington Emergency Management Funds	637,570
Total Allocations	**$4,020,476**

[a] *The amount shown reflects the cost to DOE of furnishing travel, food, and lodging for non-DOE participants at two Transportation External Coordination meetings. Participation in Transportation External Coordination meetings is not restricted to States and Tribes; however, it is not possible to break out State and Tribal costs separately.*

[b] *The amount shown reflects the cost to DOE of furnishing travel, food and lodging for non-DOE participants at three Urban Energy & Transportation Corporation meetings. Urban Energy & Transportation Corporation is a non-profit corporation organized primarily to address local government concerns.*

Besides funding, much of DOE's assistance is provided in the form of technical assistance, for which DOE bears the cost. Assistance may be provided through DOE's Radiological Assistance Program and under the National Contingency Plan, as well as through training, DOE sponsored meetings, informal discussions, and informational materials (DOE, 1994g).

2.7.3.2 Training Assistance to States and Tribes

State, Tribal, local personnel, and other Federal agencies participate in training programs developed by DOE for its staff and contractors. State, Tribal, and local personnel pay their own travel and per diem expenses; however, DOE bears the cost of developing and implementing the training. Available training includes:

- Hazardous Waste Transportation and Packaging Workshop which covers regulations governing transportation of radioactive materials;
- Radiological Emergency Response and Operations is offered in conjunction with the Federal Emergency Management Agency, and teaches response to and management of radiological incidents;
- Radiological Emergency Training for Local Responders was piloted during fiscal year 1994 in Wyoming and brings training directly to the states, allowing them to train in their own environment;
- Radioactive Material Response Orientation provides a 1-day introduction for response personnel;
- Advanced Radioactive Materials Transportation Accident Response is a sequel to the previous course for State, Tribal, Regional, and local emergency responders; and

- Transportation Emergency Training for Response Assistance includes several modules. The Public Affairs module, which is specifically designed to include States and Tribes, is scheduled for piloting during Fiscal Year 1995 (DOE, 1994g).

In addition, the DOE-funded Radiation Emergency Assistance Center/Training Site located in Oak Ridge, Tennessee, conducts courses in medical management of radiation emergencies. These courses include:

- Handling of Radiation Accidents by Emergency Personnel;
- Medical Planning and Care in Radiation Accidents;
- Health Physics in Radiation Accidents; and
- Occupational Health in Nuclear Facilities (REACT/TS, n.d.a.).

2.7.3.3 Transportation External Coordination/Working Group

DOE recognizes the need for ongoing partnerships with external organizations: health and safety; emergency management and response; law enforcement; technical; State, Tribal, and local government; and industrial organizations involved in radiological emergency response. This "stakeholder" involvement has been formalized in the Transportation External Coordination/Working Group.

The Transportation External Coordination/Working Group (Table 2-17) is a 35 member body of emergency management and response professional associations (DOE, 1994i). Through this group DOE looks at crosscutting transportation and emergency response issues that all DOE programs either are addressing or will address in the future. In turn, these groups are able to provide input to DOE for its decision-making process involving these issues (Holm, 1994).

2.7.3.4 Transportation Emergency Preparedness Program

The Transportation Emergency Preparedness Program helps integrate DOE's existing emergency management and response capabilities into an effective response system for transportation incidents involving DOE shipments. Through its extensive external coordination program with State, Tribal, and local agencies, DOE develops interfaces for meeting its various national response plan taskings to provide radiological monitoring and assessment technical assistance needed for transportation incidents involving radioactive materials including any possible incidents associated with a foreign research reactor spent nuclear fuel shipment.

Under the Transportation Emergency Preparedness Program Field Assistance Program, DOE provides support for emergency exercises (Table 2-18) that include State, Tribal, and local agencies through the Operations Offices (DOE, 1994g; DOE, 1994n; SSEB, 1994).

2.7.3.5 Radiological Assistance Program

The primary DOE response groups that would assist at a foreign research reactor spent nuclear fuel incident are the Radiological Assistance Program teams that operate from eight strategically located DOE Regional Coordinating Offices (Figure 2-19) around the country. These teams, upon State, Tribal, or local jurisdiction request, provide technical expertise and assistance to monitor and assess radiological hazards. Figure 2-19 displays pertinent information for contacting each regional office.

Table 2-17 Transportation External Coordination/Working Group Membership

Invited Organizations	*Guest Organizations*[c]
American Association of State Highway and Transportation Officials[a]	Emergency Services Representatives
American College of Emergency Physicians	Arizona Division of Emergency Services
AFL-CIO Transportation Trades Department[a]	City of Jacksonville, FL, Fire Department
American Indian Law Center[a]	Louisiana State Police/TESS
American Trucking Association[a]	St. Charles Parish Department of Emergency Management
Association of American Railroads	Ohio Emergency Management Agency
Chemical Manufacturers Association[b]	County Representatives
Columbia River Inter-tribal Fish Commission	Transportation Advisor, Nuclear Waste Project, Carson City, NV
Commercial Vehicle Safety Alliance	Clark County Comprehensive Planning Department, Esmeralda, NV
Conference of Radiation Control Program Directors	Transportation Planner, Nuclear Waste Division, Las Vegas, NV
Cooperative Hazardous Materials Enforcement Development[a]	Mineral County Office of Nuclear Projects, Hawthorne, NV
Council of Energy Resource Tribes	Nye County, NV
Council of State Governments, Midwestern Office	Nye County Nuclear Waste Repository Office, Tonopah, NV
Edison Electric Institute	White Pine County, NV
Emergency Nurses Association	White Pine County Nuclear Waste Project Office, Ely, NV
Hazardous Materials Advisory Council[a]	Tribal Representatives
International Association of Chiefs of Police	Manager, ERWM Program, Nez Perce Tribe, Lapwai, ID
International Association of Fire Chiefs	Industry Representatives
International Association of Fire Fighters	Union Pacific Railroad
International City Management Association[a]	Environmental Evaluation Group
National Association of Counties	PIC
National Association of Emergency Medical Technicians	
National Association of Regulatory Utility Commissioners[a]	
National Conference of State Legislators	
National Conference of State Transportation Specialists[a]	
National Congress of American Indians	
National Coordinating Council on Emergency Management	
National Emergency Management Association	
National Governors' Association[a]	
National Indian Policy Center	
National League of Cities[a]	
Southern States Energy Board	
Urban Energy and Transportation Corporation	
Western Governors' Association	
Western Interstate Energy Board	

[a] *Denotes organizations invited to attend, but have not yet participated.*

[b] *Denotes organizations invited to attend, but do not wish to participate.*

[c] *Denotes organizations who have attended Transportation External Coordination/Working Group meetings, but are not full-time members.*

Table 2-18 Radiological Emergency Response Exercises

Exercise	Location	Date
TRANSAX 1994	Ontario, OR	August 3, 1994
TRANSAX 1993	Lamy, NM	September 1, 1993
TRANSAX 1992	Fort Hall, ID	September 16, 1992
TRANSAX 1990	Colorado Springs, CO	November 8, 1990
WIPPTREX 93-1	Laramie, WY	April 14, 1993
WIPPTREX 92-1	Raton, NM	October 28, 1992

Radiological Assistance Program teams are composed of a range of technical specialists who volunteer for team membership. The teams are activated on an "as needed" basis and generally can be ready to deploy from their home station within 4 hours of notification. Response times to the scene vary depending on the accident's location and the level of assistance required (Taylor, 1995).

In 1994, Radiological Assistance Program teams responded 37 times to a variety of radiological situations throughout the country. Upon evaluation, a number of these responses were determined to be nonradiological hazards (Taylor and Hauptman, 1994).

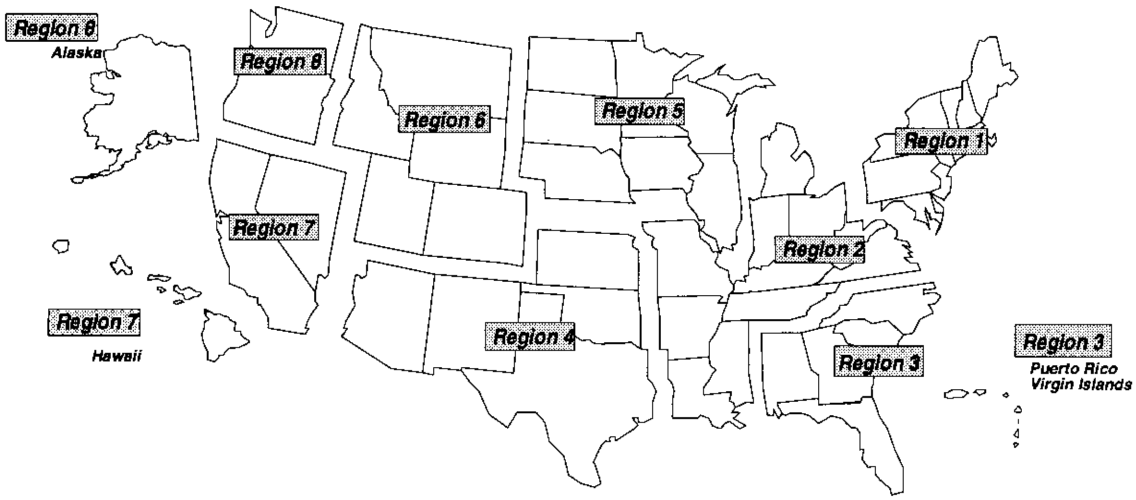
Typically, Radiological Assistance Program teams are involved in identifying personnel, equipment, or property that may be radiologically contaminated, recommending sources of medical advice for the treatment of personal injuries sustained as a result of exposure to radioactivity, and providing advice or assistance in monitoring, decontamination, material recovery, or other post-emergency operations (Gordon-Hagerty, 1993).

2.7.4 Emergency Management and Response at Ports of Entry

From 1979 to 1992, 317 spent nuclear fuel shipments in "Type B" transportation casks were made through various United States ports (NRC, 1993) with no releases of radioactive materials. The "Type B" cask shipments were placed in standard maritime shipping containers the same as any other material being sent to the port. Foreign research reactor spent nuclear fuel shipments are subject to the same types of potential hazards as those of other ships carrying nonradioactive hazardous materials.

Under the Oil Spill Prevention Act of 1990, each port is required to develop an Area Contingency Plan. While the main focus of these plans is an oil spill response, they have been expanded in many cases to address other types of hazardous material responses including radioactive material. These plans outline response capabilities, procedures, and authorities for responding to and recovering from hazardous material incidents.

These ports of entry have a specially designated and prepared terminal or dock area for unloading hazardous materials. They have either a dedicated hazardous material response team or access to a local team through some type of mutual aid agreement. These emergency response teams receive ongoing training and participate in various types of drills and exercises. Also, the dock workers receive varying levels of ongoing hazardous materials response training.



Regional Coordinating Office	Mailing Address	Telephone for Assistance
1. Brookhaven National Laboratory	Upton, Long Island New York 11973	(516) 282-2200 24 Hours
2. Oak Ridge Operations Office	DOE EOC M/S DP 801, P.O. Box 2001 200 Administration Road Oak Ridge, TN 37831-8543	(615) 576-1005 24 Hours
3. Savannah River Operations Office	P.O. Box 616 Aiken, SC 29808	(803) 725-3333 24 Hours
4. Albuquerque Operations Office	P.O. Box 5400 Albuquerque, NM 87185-5400	(505) 845-4667 24 Hours
5. Chicago Operations Office	9800 S. Cass Avenue Argonne, IL 60439	Duty Hours: (708) 252-4800 Off Hours: (708) 252-5731
6. Idaho Operations Office	850 Energy Drive Idaho Falls, ID 83401-1563	(208) 526-1515 24 Hours
7. Oakland Operations Office	1301 Clay Street Oakland, CA 94612	(510) 637-1794 24 Hours
8. Richland Operations Office	825 Jadwin Avenue Richland, WA 99352	(509) 373-3800 24 Hours
9. DOE Headquarters EOC	1000 Independence Avenue Washington, DC 20585	Non-Emergency: (202) 586-8100 Emergency: (202) 586-8500

Figure 2-19 DOE Regional Coordinating Offices for Radiological Assistance and Their Geographical Areas of Responsibility

The U.S. Coast Guard Captain of the Port would have primary On-Scene Coordinator responsibility during a foreign research reactor spent nuclear fuel incident/accident at the port, and would work in conjunction with emergency responders from the Port Authority and local jurisdictions. The On-Scene Coordinator also would be able to call upon a wide range of U.S. Coast Guard resources and the resources of other Federal agencies.

On-Scene Coordinators have at their disposal the resources of the staff of the Marine Safety Office, the resources of the staff at U.S. Coast Guard Headquarters in Washington, DC, assets of any Boat Stations in the Marine Safety Office zone, and any Air Groups. The Marine Safety Office or a Port Authority facility is used as the Emergency Operations Center for many incidents. The On-Scene Coordinator has the authority to call in the Strike Team.

These Strike Teams have limited responsibilities in the course of an incident. Their two main duties are containment and clean-up. They use booms, skimmers, absorbents, and chemicals in their response. The vessels used as platforms for booms and skimmers are usually provided by the Marine Safety Office or Boat Station in the area of the incident. Strike Teams consist of highly trained pollution response and clean-up personnel.

There are three Strike Teams under the command of the National Strike Force Coordination Center in Elizabeth City, NC, that could be called on for a foreign research reactor spent nuclear fuel accident. These teams are located on the three coasts of the United States: the Gulf Strike Team located in Mobile, AL; the Pacific Strike Team located in Novato, CA; and the Atlantic Strike Team located in Fort Dix, NJ.

For a foreign research reactor spent nuclear fuel accident, the U.S. Coast Guard On-Scene Coordinator would request an NRC or DOE representative. DOE Radiological Assistance Program teams would be requested as needed.

2.7.5 Emergency Management and Response Along Ground Transport Routes

During transport of the spent nuclear fuel received as a result of the Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel Environmental Assessment, the foreign research reactor operator's shipping agents were required to ensure that all activities of the agent's and the commercial carrier's personnel conformed to regulatory requirements, as well as all plans and procedures developed for the foreign research reactor spent nuclear fuel (DOE, 1994m). DOE and other Federal and State government agencies monitored the activities of the shipping agent and the commercial carrier to ensure that the regulatory requirements were met. If foreign research reactor spent nuclear fuel is managed in the United States, DOE will prepare a Transportation Plan before any shipments are undertaken. The Transportation Plan will detail all transportation activities necessary for the safe and secure transport of the foreign research reactor spent nuclear fuel from the point of origin to the management site in the United States. The general provisions for such a plan are included in Appendix H.

Primary responsibility for emergency response to a foreign research reactor spent nuclear fuel incident would reside with local authorities (DOE, 1989). Each corridor State or Tribe would be responsible for augmenting their existing emergency management and response plans and procedures with any foreign research reactor spent nuclear fuel specific information they felt was necessary.

States coordinate with their local jurisdictions on emergency planning and information. States and Tribes would be responsible for notifying DOE of any conditions that could affect the safe and secure transport of foreign research reactor spent nuclear fuel shipments through their jurisdictions. DOE would provide technical advice and assistance to the shippers and affected government jurisdictions to ensure safe transportation.

Spent nuclear fuel shipments transported by rail, barge, or commercial truck carrier would be subject to the same potential problems as any other hazardous materials shipment that travels daily by these means, namely severe weather, mechanical problems, derailments, and collisions.

DOE would seek to mitigate potential highway foreign research reactor spent nuclear fuel accident consequences by ensuring commercial carriers comply with NRC guidelines for shipment security and the Department of Transportation Highway Route Controlled Quantity routing regulations (DOE, 1995c) which are designed to reduce radiological transportation risk impacts.

The rail and barge industries are similar to the trucking industry in the hazardous material transportation regulatory regime. Documentation, manifesting, placarding, labeling, and other communications are controlled by 49 CFR Parts 100-199.

The carriers used for the transportation of foreign research reactor spent nuclear fuel would be required to develop emergency response plans. In developing these plans, the carriers would be required to consider the following responsibilities:

- protect life, health, and the environment;
- notify appropriate railroad officials in a timely manner;
- notify the appropriate Federal, State, and local authorities, and the shipper;
- initiate a prompt and proper response;
- provide appropriate resources and expertise for resolution of the incident;
- perform cleanup functions; and
- establish and maintain a working contact with the responsible Governmental authorities until they declare the incident closed.

As discussed in Section 2.7.2, spent nuclear fuel shipments, like other hazardous material shipments, have been involved in transportation accidents. Those that have occurred have not resulted in a radiological hazard or damage to the public or the environment. This is primarily due to the rigorous packaging.

Each State and Tribe along a shipping route would be notified of the foreign research reactor spent nuclear fuel shipment's itinerary through that jurisdiction to enable the appropriate agencies to notify the necessary response personnel. Also, DOE would maintain continuous communications through its communications and tracking systems (DOE, 1989).

Foreign research reactor spent nuclear fuel shipments would be tracked either by the commercial carrier or by a satellite tracking system similar to DOE's Transportation Tracking and Communications System (Figure 2-20). The satellite tracking system would provide a "real-time" satellite tracking and voice communications system that would link the truck or train and its escorts with a control center. Some commercial carriers have established their own satellite tracking systems. The DOE system would interface with these systems and co-monitor the shipment's progress to ensure maximum accountability and security. The satellite tracking system would also coordinate "SAFE PARKING" requests from the states.

If a situation would arise (e.g., severe weather, mechanical difficulties, protesters, security threat, personnel illness or injury) that presented a hazard or threat to a highway foreign research reactor spent nuclear fuel shipment, DOE would have arranged through Memoranda of Agreement for the commercial carrier to divert to any Federal installation (e.g., a DOE site or military base) and request "SAFE PARKING" at that facility until the situation is resolved. The receiving facility would assist in providing security and logistical support until the shipment was prepared to depart.

State, Tribal, and local agencies, as well as the commercial carriers, maintain various emergency response plans and procedures. During a foreign research reactor spent nuclear fuel highway, barge, or rail accident, the personnel accompanying the shipment would be the immediate contact for information to the local emergency responders having jurisdiction and Incident Commander authority over the situation.

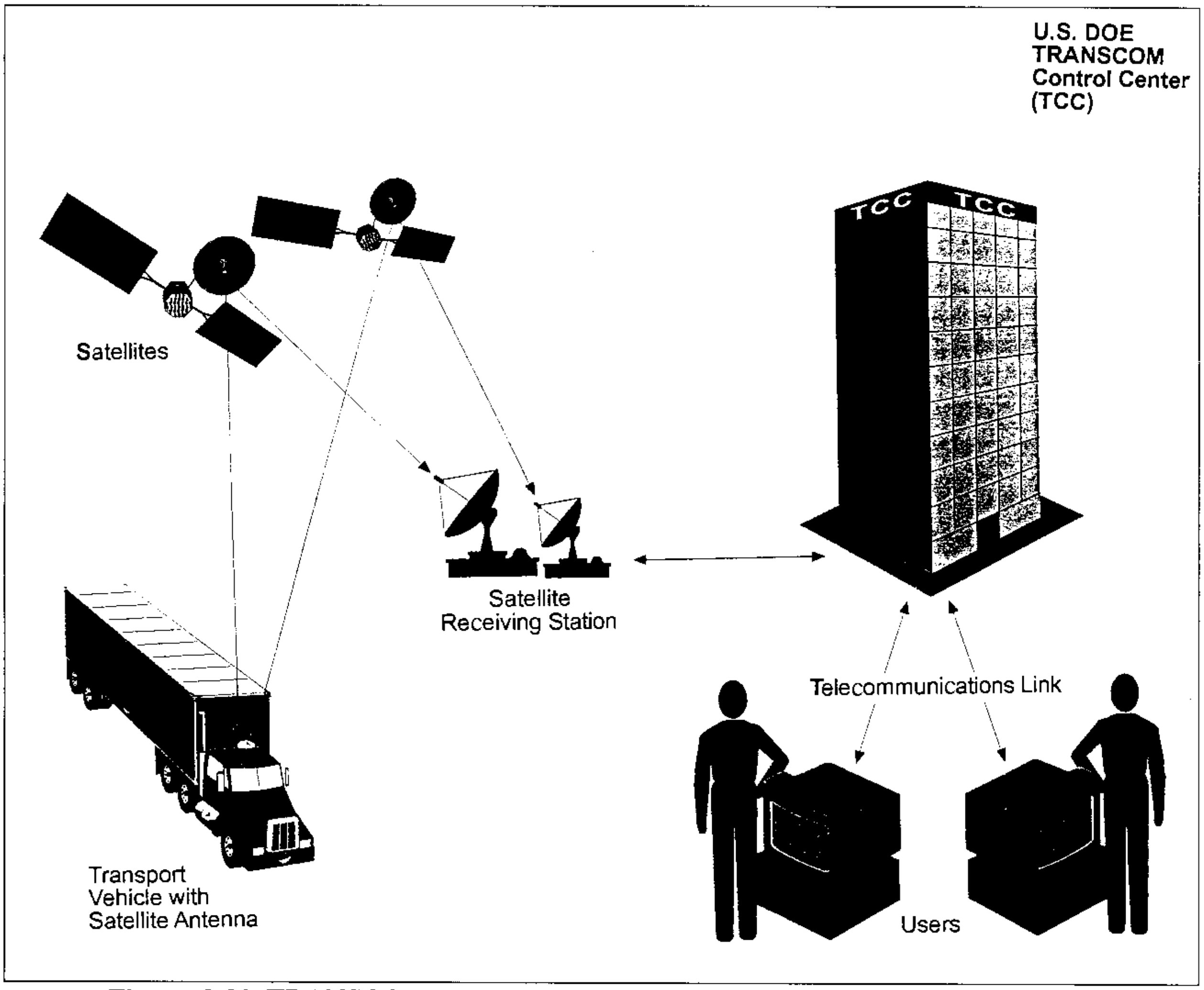


Figure 2-20 TRANSCOM, DOE's Transportation Tracking and Communications System

Additionally, the Hazardous Material Regulations (49 CFR 177.861) advise highway shippers, carriers, and emergency responders to contact DOE if assistance with radioactive materials is required (DOE, 1990b). A DOE Radiological Assistance Program team could respond to the scene if requested.

Incident Commanders have other sources of technical assistance they could call on such as the commercial carrier's technical experts (through a 24-hr contact number), the National Response Center, and the Chemical Transportation Emergency Center, which provides immediate response advice and information from the shipper on a 24-hr basis.

2.7.6 Emergency Management and Response at Management Sites

The DOE 5500 series of Orders incorporates various Federal regulatory requirements and mandates an extensive emergency management and response program at each management site in the same manner as any other industrial facility or local government jurisdiction. State-level specific requirements are addressed by each respective site.

Each of these sites routinely handles hazardous materials that have potential emergency management and response considerations similar to foreign research reactor spent nuclear fuel. These, along with the usual risks posed by any industrial environment, are regularly evaluated through various Safety Analysis Reports and Hazards Assessment studies. These situations are then mitigated to the greatest extent possible.

2.8 Security Measures

Domestic transportation of foreign research reactor spent nuclear fuel would be under the regulatory jurisdiction of the U.S. Department of Transportation and NRC. In the event that foreign research reactor spent nuclear fuel were to be transported through a military port of entry, applicable requirements would be established in advance by the U.S. Department of Defense, DOE and NRC to provide the appropriate level of security.

The objective of the security measures during transportation of spent nuclear fuel are to minimize the possibilities for radiological sabotage of spent nuclear fuel shipments and facilitate the location and recovery of spent nuclear fuel shipments that may have come under control of unauthorized persons. The elements of the security measures would be considered when developing the Transportation Plan to be developed by DOE in consultation with State, local, and Tribal officials prior to any actual spent nuclear fuel shipments. The general provisions of the Transportation Plan, which would include requirements relative to emergency response planning, security considerations, and communications during actual shipments of foreign research reactor spent nuclear fuel, are included in Appendix H.

The security measures provided by the regulations would make the hijacking of a transportation cask a highly unlikely event. In the first place, the large size and weight of these casks (9.1 to 22.7 metric tons, or 10 to 25 tons) and the inherent radioactivity of the spent nuclear fuel make spent fuel in a transportation cask an unlikely hijacking target. For a malicious act of sabotage, there are, in fact, more accessible targets than spent nuclear fuel, that would provide more spectacular detrimental effects; especially considering the fact that, aside from the radioactivity of the spent nuclear fuel, which is a relatively short range effect, the spent nuclear fuel elements are simply pieces of metal (which might be somewhat warm). In the event of a hijack attempt aimed at some long-term use of the contents of the cask, the communications systems required to be used during the shipment would enable timely notification of authorities who would mobilize response forces. Tracking systems would allow the location of the cask to be determined in real time, thereby aiding in the timely interception of the hijackers by response forces.

The successful completion of attempts aimed at short-term destructive acts, such as explosions from within the cask or inducement of criticality, are not considered credible because they would require sufficient time to breach the cask at a great personal risk to the hijackers (probably lethal exposure), special tooling and techniques, and/or the use of specialized materials (for sufficient moderation) that in themselves are safeguarded materials.

Malicious attack scenarios from a distance, such as the explosion of a bomb near a transportation cask, or an attack by an armor-piercing weapon could be within the realm of possibility. The risk to the health and safety of the public associated with such an event cannot be calculated since there is no basis for estimating either the probability of such an event occurring or that damage sufficient to release radioactive material from the cask would occur. Appendix D, Section D.5.9, provides a discussion of the consequences of some sabotage/terrorist initiated events for the purpose of emergency response planning.

2.9 Preferred Alternative

In selecting a preferred alternative for the management of foreign research reactor spent nuclear fuel, DOE and the Department of State took several factors into consideration, including the following:

1. U.S. Government nuclear weapons nonproliferation policies and objectives;
2. DOE responsibilities (e.g., safe handling of hazardous materials, safety/health risks to workers, compatibility with other ongoing missions, etc.);
3. Potential environmental impacts (e.g., public safety, etc.);
4. Public comments received and concerns expressed following issuance of the Draft EIS;
5. Analysis of impacts and alternatives in the Programmatic SNF&INEL Final EIS (DOE, 1995c), as well as the Record of Decision for that EIS;
6. Estimated costs of alternatives for management of foreign research reactor spent nuclear fuel;
7. Public issues/concerns/perceptions (e.g., fairness/equity to affected States and populations, etc.); and
8. Uncertainties (e.g., future budget priorities and continuity of funding, technology development, repository timing and waste form acceptance criteria, regulatory change, etc.).

Based on consideration of these factors, DOE and the Department of State, in consultation with other Government agencies, designate the alternative described below as the preferred alternative. This preferred alternative is the same as Management Alternative 1 (Manage Foreign Research Reactor Spent Nuclear Fuel in the United States, discussed in Section 2.2), with the modifications discussed below. The basic components of Management Alternative 1 have been modified to incorporate various implementation alternatives discussed in Section 2.2.2.

The amount of foreign research reactor spent nuclear fuel that would be accepted and managed, as specified in Section 2.2.1.3, could total approximately 19.2 MTHM, with a volume of approximately 110 m^3 (4,100 ft^3), representing approximately 22,700 individual spent nuclear fuel elements. The target material that would be accepted and managed, as specified in Section 2.2.2.1, contains an additional 0.6 MTHM representing the uranium content of approximately 620 additional typical foreign research reactor spent nuclear fuel elements. The following stipulations on qualifying spent nuclear fuel types would apply:

- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective.
- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors which operate on HEU fuel when the policy becomes effective and which agree to convert to LEU fuel. Spent nuclear fuel would not be accepted from research reactors that could convert to LEU fuel but refuse to do so.

- Spent nuclear fuel (HEU) would be accepted from research reactors having lifetime cores, from research reactors planning to shut down by a specific date while the policy is in effect, and from research reactors for which a suitable LEU fuel is not available.
- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors that are already shut down.
- Unirradiated fuel (HEU and/or LEU) from eligible research reactors would be accepted as spent nuclear fuel.
- For research reactors with both HEU and LEU spent nuclear fuel available for shipment, LEU spent nuclear fuel would not be accepted until the HEU spent nuclear fuel is exhausted, unless there are extenuating circumstances (e.g., deterioration of one or more LEU elements sufficient to cause a safety problem).
- Spent nuclear fuel (HEU and/or LEU) would not be accepted from new research reactors starting operation after the date of implementation of the policy.

The policy duration under this preferred alternative would be 10 years, beginning on the date when the management policy would become effective, as discussed in Section 2.2.1.1. Shipments of spent nuclear fuel to the United States could be made for a period of 13 years, starting from the effective date of policy implementation, as long as the spent nuclear fuel had already been discharged prior to the beginning of the policy period or is discharged during the policy period.

The aluminum-based foreign research reactor spent nuclear fuel would be managed at the Savannah River Site and the TRIGA foreign research reactor spent nuclear fuel would be managed at the Idaho National Engineering Laboratory, in accordance with the Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995c) and the settlement agreement reached between DOE and the State of Idaho [Public Service Co. of Colorado v. Batt, No. CV 91-0035-S-EJL (D. Id.) and United States v. Batt, No. CV-91-0054-S-EJL (D. Id.)]. Under this preferred alternative, up to approximately 19 MTHM of aluminum-based foreign research reactor spent nuclear fuel (approximately 17,800 elements), representing up to approximately 675 casks, and target material representing up to approximately 140 additional casks would be accepted and managed at the Savannah River Site. Also, up to approximately 1 MTHM of TRIGA foreign research reactor spent nuclear fuel (approximately 4,900 elements), representing up to approximately 162 casks would be accepted and managed at the Idaho National Engineering Laboratory.

The candidate U.S. ports of entry are listed in Section 2.2.1.6 and are described in detail in Section 3. Although all of the ports are acceptable based on the port selection criteria discussed in Appendix D, DOE would prefer to candidate use the military ports in proximity to the spent nuclear fuel management sites (i.e., Charleston NWS and the Concord NWS). Under this preferred alternative, a maximum of 38 casks of TRIGA foreign research reactor spent nuclear fuel (estimated to require about 5 shipments) could be accepted at a Western port, with 150 to 300 shipments being accepted via an Eastern port.

The foreign research reactor spent nuclear fuel and target material would be shipped by either chartered or regularly scheduled commercial ships from the foreign ports to the United States, as specified in Section 2.2.1.5.

DOE would take title to the foreign research reactor spent nuclear fuel and target material that is shipped by sea after it is offloaded at the port of entry, and to the spent nuclear fuel and target material shipped solely overland (i.e., from Canada) at the border crossing between Canada and the United States.

The foreign research reactor spent nuclear fuel and target material would be transported from the United States ports to the management sites by truck and rail as specified in Section 2.2.1.7.

The financing arrangement under this preferred alternative would be for the United States to bear the full cost for transporting and managing the foreign research reactor spent nuclear fuel and target material accepted from countries with other-than-high-income economies, and to charge high-income economy countries a competitive fee. The fee would be established in a Federal Register Notice (as opposed to being published in this Final EIS), to allow DOE flexibility to adjust the fee to account for inflation, or changes in spent nuclear fuel management practices in the United States.

For the aluminum-based foreign research reactor spent nuclear fuel, a three point strategy is proposed, as follows:

1. DOE would embark immediately on an accelerated program at the Savannah River Site to identify, develop, and demonstrate one or more non-reprocessing, cost-effective treatment and/or packaging technologies to address potential health and safety issues that may develop and to prepare the foreign research reactor spent nuclear fuel for ultimate disposal. The purpose of any new facilities that might be constructed to implement these technologies would be to change the foreign research reactor spent nuclear fuel into a form that is suitable for geologic disposal, without necessarily separating the fissile materials, while meeting or exceeding all applicable safety and environmental requirements. Examples of technologies that would be considered include: *can-in-canister, chop and dilute/poison, melt and dilute/poison, plasma arc treatment, electrometallurgical treatment, glass material oxidation and dissolution, chloride volatility, dissolve and vitrify, direct disposal in small packages*, etc. Functional schematics of these technologies are shown in Figure 2-21. In conjunction with the examination of new technologies, variations of conventional direct disposal methods would also be explored. After treatment and/or packaging, the foreign research reactor spent nuclear fuel would be managed on site in "road ready" dry storage until transported off-site for continued storage or disposal. DOE would select, develop, and implement, if possible, one or more of these treatment and/or packaging technologies by the year 2000. DOE is committed to avoiding indefinite storage of this spent nuclear fuel in a form that is unsuitable for disposal.

2. Despite DOE's best efforts, it is possible that a new treatment and/or packaging technology may not be ready for implementation by the year 2000. It may become necessary, therefore, for DOE to use the F-Canyon to reprocess some foreign research reactor spent nuclear fuel elements, while the F-Canyon is operating to stabilize at-risk materials as recommended by the Defense Nuclear Facilities Safety Board. (For example, under current schedules this activity could take place between the years 2000 and 2002.) In that event, the foreign research reactor spent nuclear fuel would be converted into LEU and wastes generated during reprocessing. Certain wastes would be vitrified in the Defense Waste Processing Facility, while others would be solidified in the Saltstone facility. In order to provide a sound policy basis for making a determination on whether and how to utilize the F-Canyon for processing tasks that are not driven by health and safety considerations, DOE will commission or conduct an independent study of the nonproliferation and other (e.g., cost and timing) implications of reprocessing spent nuclear fuel from foreign research reactors. The study will be initiated in mid-1996 and will be completed in a timely fashion to allow a subsequent decision about possible use of the F-Canyon for foreign research reactor spent nuclear fuel reprocessing to be fully considered by the public, the Congress and the Executive Branch agencies. Pending disposition of the foreign research reactor spent

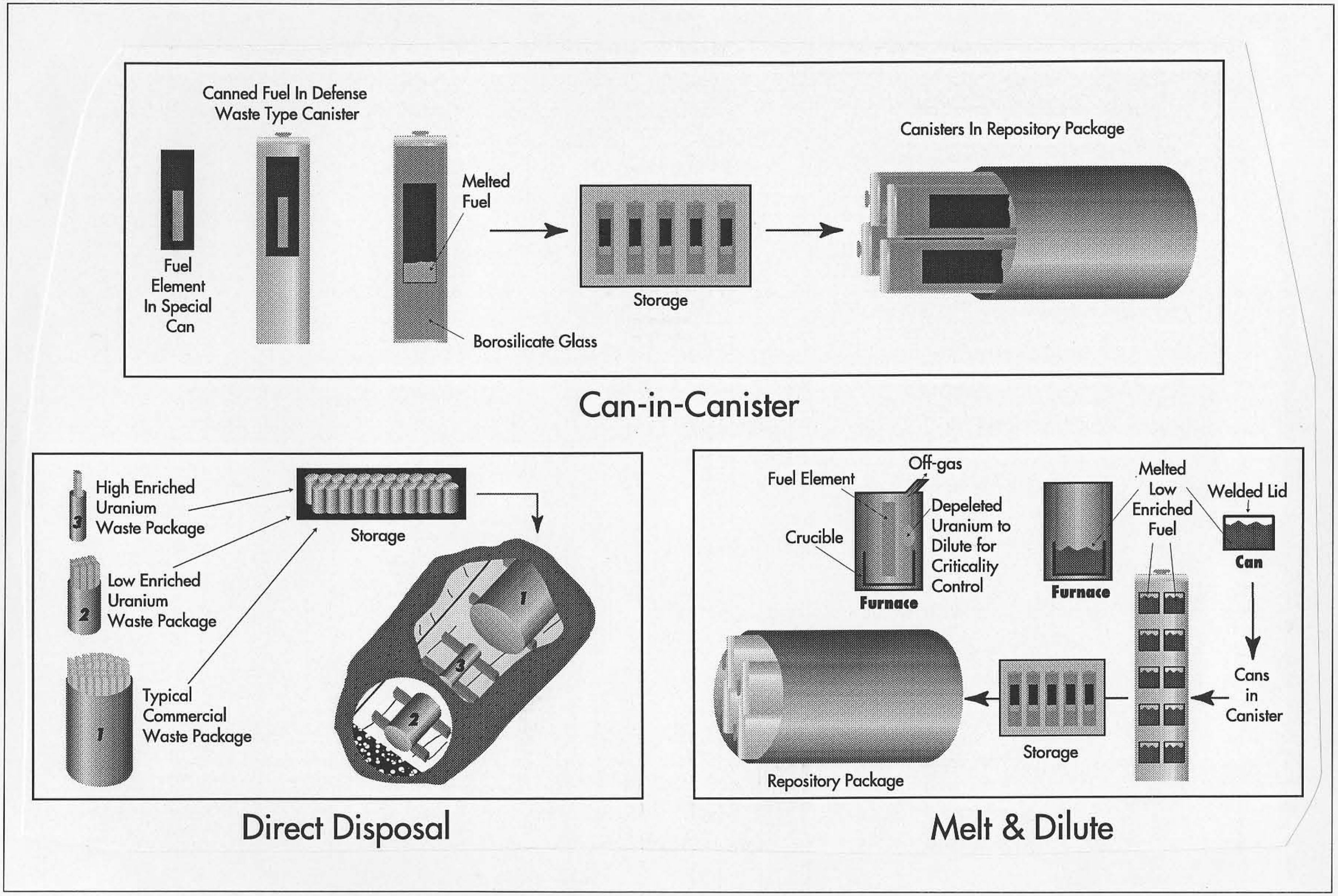


Figure 2-21 New Treatment and Packaging Technologies (Functional Schematic Diagrams)

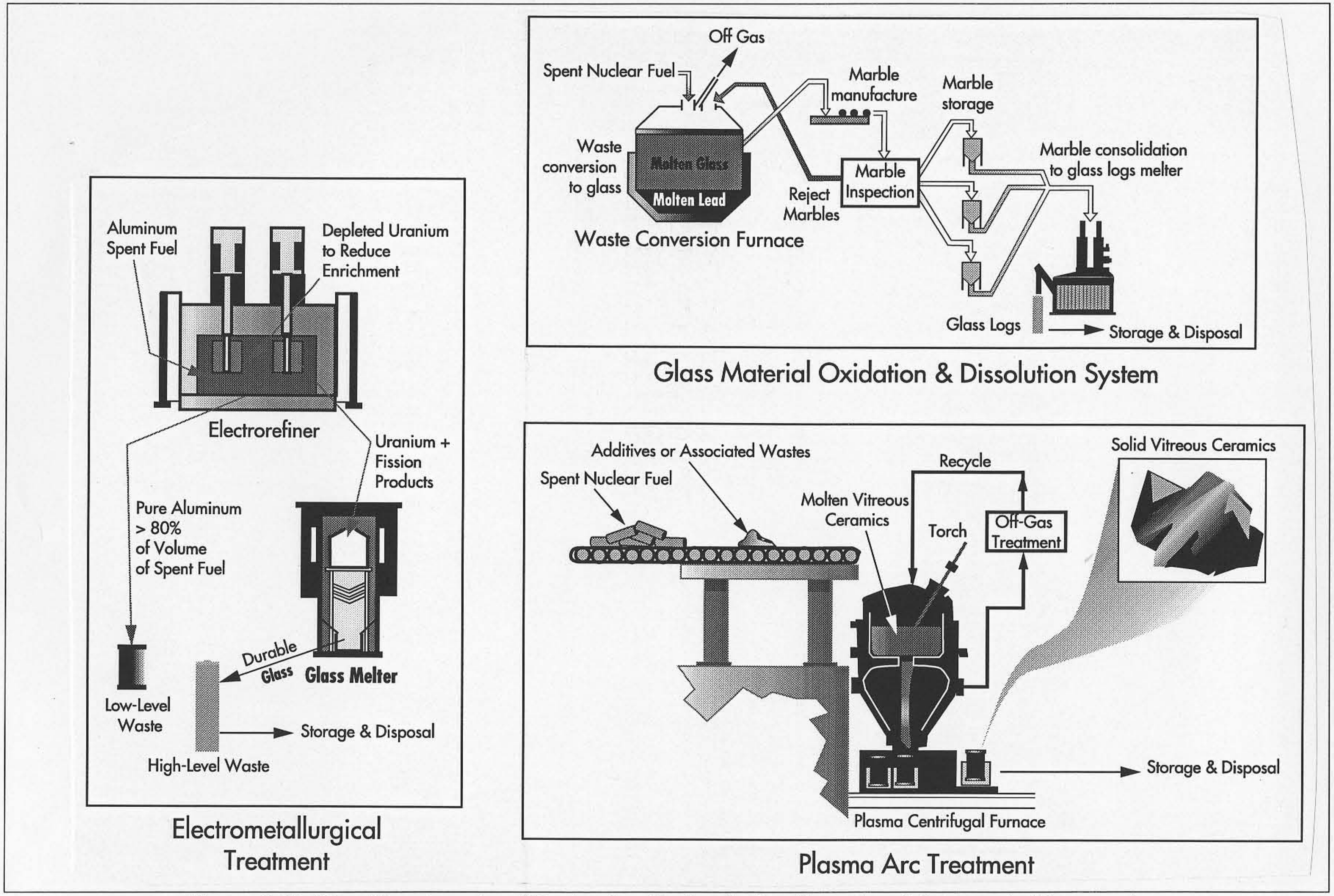


Figure 2-21 New Treatment and Packaging Technologies (Functional Schematic Diagrams) (Continued)

nuclear fuel by either a new treatment and/or packaging technology or reprocessing in the F-Canyon, the spent nuclear fuel would be placed in existing wet storage at the Savannah River Site.

3. DOE would conduct a program of close monitoring of any foreign research reactor spent nuclear fuel and target material that would be accepted for storage in existing wet storage facilities. DOE is presently unaware of any technical basis for believing that this spent nuclear fuel cannot be safely stored until one or more of the treatment and/or packaging technologies becomes available. Nevertheless, if health and safety concerns involving any of the foreign research reactor spent nuclear fuel elements are identified prior to development of an appropriate treatment and/or packaging technology, DOE would use the F-Canyon to reprocess the affected spent nuclear fuel elements, if it is still operating to stabilize at-risk materials.

   Because of criticality constraints stemming from the configuration of the F-Canyon, under no circumstances would it be possible to produce separated HEU that is suitable for a nuclear weapon. Instead, depleted uranium would be added to the foreign research reactor spent nuclear fuel near the beginning of the reprocessing process, so that only LEU would be produced when the uranium is separated from the fission products. The trace quantities of plutonium in the spent nuclear fuel would be left in and solidified along with the high-level radioactive reprocessing wastes. This would further the President's policy to discourage the accumulation of excess weapons grade fissile materials, to strengthen controls and constraints on these materials and, over time, to reduce worldwide stocks.

The TRIGA foreign research reactor spent nuclear fuel would be stored at the Idaho National Engineering Laboratory in the Fluorinel Dissolution and Fuel Storage (FAST) facility (wet storage) or preferably the dry storage Irradiated Fuel Storage Facility (IFSF) and the CPP-749 dry storage area. After 2003, all foreign research reactor spent nuclear fuel would be managed in accordance with the provisions of the settlement agreement between DOE and the State of Idaho, until transported off-site for ultimate disposition. Depending on the nature of any new treatment and/or packaging technology that might be developed, the TRIGA spent nuclear fuel would also be processed using such a new technology, if necessary for disposal.

A critical result of implementing this preferred alternative would be the continued viability and vitality of the Reduced Enrichment for Research and Test Reactors (RERTR) Program, whose goal is minimizing and eventually eliminating the use of HEU in civil nuclear programs, by providing foreign research reactor operators with a continued incentive to participate. Similarly, the successful development of alternative fuels for research reactors and the expansion of the program to Russia, the other Newly Independent States, China, South Africa, and other countries, and the establishment of a world-wide norm discouraging the use of HEU, are dependent on the United States' commitment to action such as that embodied in this preferred alternative.

DOE is aware that the inclusion of chemical separation within the preferred alternative could be interpreted by some nations, organizations and persons as a signal of endorsement of the use of reprocessing as a routine method of waste management for spent nuclear fuel or a reversal of U.S. policy on reprocessing. This would not be an accurate interpretation. The U.S. policy regarding reprocessing was established in Presidential Decision Directive 13. DOE and the Department of State have determined that this preferred alternative is not inconsistent with that policy. The draft version of this EIS indicated that reprocessing is a non-preferred technology and would not be used unless one or more of a set of specific conditions occurred (see Draft EIS Section 2.2.2.6). This final preferred alternative, which includes reprocessing, establishes a prescribed set of circumstances that would have to be met before reprocessing would be used. The independent study discussed above in point 2 of the strategy for management of aluminum-based spent nuclear fuel will review the policy, technology, cost and schedule implications for reprocessing foreign research reactor spent nuclear fuel to determine whether reprocessing of foreign research reactor spent nuclear fuel is justified for other than health and safety reasons.

Policy considerations and environmental impacts associated with implementation of this preferred alternative are presented in Section 4.7. Cost considerations are included in Section 4.9.

Basis for the Preferred Alternative - The elements of the preferred alternative discussed above have been selected based on the following considerations:

1. ***Management Alternative*** - The various management alternatives considered are discussed in Sections 2.2 through 2.4 of the EIS. The analyses in Sections 4.2 through 4.5 of the EIS demonstrate that the impacts on the environment, involved workers, or the citizens of the United States from implementation of any of the management alternatives or implementation alternatives analyzed (other than beneficial impacts associated with support for United States nuclear weapons nonproliferation policy) would be small and completely within the applicable regulatory limits, and would not provide a basis for discrimination among the alternatives. As a result, the process for selection of the elements of the preferred alternative focused on programmatic considerations:

    a. DOE and the Department of State concluded that the No Action Alternative and Management Alternative 2, Implementation Alternative 1a (Overseas Storage) would be unacceptable since these alternatives are not consistent with United States nuclear weapons nonproliferation policy objectives.

    b. DOE and the Department of State believe that the basic implementation of Management Alternative 1 would be undesirable to the extent that it would involve indefinite storage of foreign research reactor spent nuclear fuel in a form that is not suitable for disposal. Management Alternative 1 modified to rely solely on Implementation Alternative 6 (Near Term Conventional Chemical Separation in the United States) would raise nuclear weapons nonproliferation policy questions. Management Alternative 1 modified to rely solely on Implementation Alternative 7 (Developmental Treatment and/or Packaging Technologies) could not be selected at this time because no decision has been made on which technology will be pursued.

    c. DOE and the Department of State also believe that Management Alternative 2, Implementation Alternative 1b (Overseas Reprocessing) would be technically complex and potentially extremely expensive because it would require the United States to accept reprocessing wastes from the overseas reprocessing operations. This is due to the fact that both of the countries in which the overseas reprocessing might be accomplished require the reprocessing wastes to leave their countries, and many of the countries that would be covered by the proposed policy cannot accept the return of such reprocessing wastes. The intermediate-level radioactive wastes produced in Europe during reprocessing of research reactor spent nuclear fuel are often in a concreted waste form, unlike any high-level radioactive waste form in the United States. This concreted waste form has not been evaluated for disposal in a United

States geologic repository. Accordingly, acceptance of such waste in the United States likely could require expensive, currently unproven treatment and/or packaging technologies to transform it into a form that would be acceptable for disposal.

d. The sample hybrid alternative (Management Alternative 3) analyzed in the Draft EIS involved partial reprocessing overseas coupled with partial management in the United States. In order for this alternative to be consistent with United States nuclear weapons nonproliferation policy objectives, certain conditions would have to be met by either the reprocessor (e.g., Dounreay) or the research reactor operators. Staff from both DOE and the Department of State have addressed this issue with representatives of the United Kingdom Department of Trade and Industry and reactor operators, and have determined that it would not be possible to ensure compliance with the United States nuclear weapons nonproliferation policy objectives. The primary concern was the inability to ensure that any separated HEU would be blended down to LEU. Obtaining the reactor operators' agreement to such a policy would likely require significant financial subsidies. The potential cost of achieving agreement to blend down the uranium, plus uncertainties regarding Dounreay's long-term availability, led DOE and the Department of State to conclude that successful implementation of this alternative could not be relied on.

None of the alternatives analyzed in the Draft EIS could be implemented without some degree of difficulty. However, a modification of Management Alternative 1 (Manage Foreign Research Reactor Spent Nuclear Fuel in the United States), incorporating a combination of alternatives to the basic implementation components balances policy, technical, cost and schedule requirements. DOE and the Department of State consider that this approach provides the highest assurance that programmatic requirements could be met. This combination also provides the strongest support for United States nuclear weapons nonproliferation policy objectives as all aspects of the alternative would be under the control of DOE, either directly or through the spent nuclear fuel acceptance contracts with the reactor operators.

2. ***Management Technology*** - The alternative spent nuclear fuel management technologies considered are discussed in Sections 2.2.2.7 and 2.6.5 of the EIS. The approaches fall into four broad categories, as follows:

Wet Storage - Wet storage is a proven technology, the impacts of which would be small, and completely within the applicable regulatory limits, if it were used to implement the proposed action. Furthermore, DOE currently has wet storage facilities in operation at the Savannah River Site and the Idaho National Engineering Laboratory that could be used for storage of foreign research reactor spent nuclear fuel. However, wet storage requires attention to ensure that the storage conditions do not foster slow degradation of the spent nuclear fuel through corrosion.

Dry Storage - Dry storage is also a proven technology, that would also have no more than small impacts, completely within the applicable regulatory limits, if used to implement the proposed action. It is the storage medium that is being selected at all commercial power reactor sites where additional storage capacity is being built. However, it has not been used for research reactor spent nuclear fuel in the United States. Dry storage capacity could be provided at the management sites in time to meet the program's projected needs, if initial spent nuclear fuel receipts were placed into the available wet storage.

Chemical Separation - Chemical separation is also a proven technology, the impacts of which would be small, and completely within the applicable regulatory limits, if used to implement the proposed action. However, DOE is phasing out its chemical separation activities and is currently reprocessing only at the Savannah River Site to stabilize materials for health and safety reasons. Because these chemical separations facilities could be used to treat the foreign research reactor spent nuclear fuel, they provide a contingency to be considered pending availability of an alternate means of treating and/or packaging the spent nuclear fuel prior to ultimate disposition.

New Technologies - Due to concerns regarding geologic disposal of intact spent fuel containing HEU (i.e., the possibility of uncontrolled criticality incidents), some form of treatment of this spent nuclear fuel may be required. While several concepts have been proposed for new treatment and/or packaging technologies, none of them are ready for implementation at this time. Prior to a decision leading to their implementation, additional development work would be required to determine whether and how they could be implemented, based on technical and cost considerations.

In order to effectively implement the preferred alternative of accepting and managing the foreign research reactor spent nuclear fuel in the United States, DOE and the Department of State developed the three point strategy for management of aluminum-based spent nuclear fuel discussed earlier in this Section. This strategy draws on the strengths of each of the spent nuclear fuel management technologies discussed above, while avoiding sole reliance on any of them. Due to the relatively more robust nature of the TRIGA spent nuclear fuel, DOE believes that minimal additional development may be needed to prepare it for storage and final disposition. Accordingly, the preferred alternative specifies that the TRIGA spent nuclear fuel would be placed in existing dry storage facilities at the Idaho National Engineering Laboratory. However, the program to qualify the final geologic disposal form for the TRIGA spent nuclear fuel will continue and the appropriate treatment, if any, would be identified and implemented.

3. ***Policy Duration*** - The alternative policy durations considered are defined in Sections 2.2.2.1 and 2.2.2.2 of the EIS. Analysis of these alternatives concluded that the 5-year option is likely to provide insufficient time for the reactor operators to arrange for alternative spent nuclear fuel disposal mechanisms, and thus might result in some reactor operators refusing to cooperate fully with United States nuclear weapons nonproliferation programs. This, in turn, could undermine international cooperation with other nuclear weapons nonproliferation programs the United States might seek to implement.

   On the other hand, the analysis determined that there was insufficient benefit to be gained from indefinite acceptance of all of the spent nuclear fuel containing HEU because such an approach likely would provide insufficient incentive for other countries to proceed expeditiously with arrangements for alternative disposal mechanisms not involving the United States.

   The approach incorporated into the preferred alternative allows sufficient incentive to the reactor operators to ensure their cooperation, while specifying a definite cut-off point. This alternative provides sufficient lead time to allow the reactor operators to make other arrangements for disposition of their spent nuclear fuel, and provides sufficient time to accept all spent nuclear fuel containing HEU enriched in the United States.

4. ***Amount of Material to Manage*** - The alternative amounts of material that might be covered by the proposed policy are defined in Sections 2.2.1.3 and 2.2.2.1 of the EIS. DOE and the Department of State concluded that management of spent nuclear fuel only from other-than-high-income economy countries would strongly encourage the resurgence of the use of HEU in the high-income economy countries, as well as opening the United States, fairly or unfairly, to charges that we are not living up to our commitments under the *Treaty on the Non-Proliferation of Nuclear Weapons*. Management of only spent nuclear fuel containing HEU would penalize those reactors that have already converted to the use of LEU fuel, and would provide an incentive for reactors to continue to use HEU fuel, or switch back to its use. The impacts that would result from management of any of these different amounts of material would be small, and within the applicable regulatory limits.

   DOE and the Department of State concluded that management of all of the aluminum-based and TRIGA foreign research reactor spent nuclear fuel currently in storage or projected to be discharged during the policy period, and target material containing uranium enriched in the United States, would provide the best support for the objectives of the proposed policy. Implementation of this preferred alternative would provide an opportunity for removal of the maximum amount of HEU from civil commerce and would provide an incentive for the continued conversion to and use of LEU as fuel for foreign research reactors, in place of highly enriched (weapons grade) uranium.

5. ***Marine Transport*** - The alternative approaches to marine transport of foreign research reactor spent nuclear fuel are discussed in Section 2.2.1.5 of the EIS. The analysis in the EIS demonstrates that the impacts to the environment, workers or the public from transport of the spent nuclear fuel using any of these types of ships would be small, and within the regulatory limits. The analyses do not identify any difference in the small impacts that would result from the use of purpose-built vs. general purpose ships. Since "military transports" are in fact the same type of ship as the chartered commercial cargo ships and are crewed by civilians, use of "military transports" would not actually result in any difference in impacts. DOE and the Department of State believe that use of actual warships would be both unnecessary from a security standpoint and could entail additional risk to the environment and the public, since such ships do not routinely carry cargo.

   The approach selected by DOE and the Department of State for the preferred alternative provides maximum flexibility for marine transport.

6. ***Ground Transport*** - The ground transportation alternatives are defined in Section 2.2.1.7 of the EIS. The analyses in the EIS demonstrate that the impacts to the environment, workers or the public, from any of these modes of ground transport (counting barge as a mode of "ground transport") would be small and within the applicable regulatory limits. Furthermore, the differences in potential impacts between the truck, rail and barge alternatives were not significant.

   Both the truck and rail transportation options have been used successfully to transport foreign research reactor spent nuclear fuel in the past. Truck transport was the predominant mode used for over twenty years, until the old "Off-Site Fuels Policy" lapsed in 1988. Rail was the mode used for both shipments under the *Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel* (DOE, 1994m). Since neither of the preferred ports of entry (see item 8 below) can reasonably provide barge transport to either of the proposed management sites, barge transport was dismissed from consideration in the preferred alternative.

By providing for either truck or rail transport, the preferred alternative would build on previous satisfactory experience while providing maximum flexibility for dealing with changes in the transportation process in the future.

7. ***Title Transfer Location*** - The alternative points at which DOE might take title to the spent nuclear fuel and target material are discussed in Sections 2.2.1.4 and 2.2.2.4 of the EIS. The point at which title would be transferred has no effect on the physical processes that would take place, and thus would not have any effect on the impacts on the environment, workers or on the public. The Price-Anderson Act would provide liability protection in the unlikely event of a nuclear accident in the United States, whether or not DOE had taken title to the spent nuclear fuel at the time of such an accident. As a result, DOE and the Department of State concluded that the selection of the title transfer location could be made solely on programmatic considerations.

   Acceptance of title at the foreign research reactor sites could make the United States Government liable for any accident that might occur in the country of origin, or on the high seas. DOE and the Department of State have been unable to identify any advantage to the United States of taking title outside the United States.

   Taking title at the limit of United States territorial waters makes the title transfer depend solely on when the ship enters United States waters, which could be difficult for DOE to control in certain circumstances (e.g., a storm).

   Acceptance of title when the foreign research reactor spent nuclear fuel actually enters the land mass of the United States provides the most certainty for implementation.

   The approach incorporated into the preferred alternative ensures that liability for accidents during the transportation process outside the United States would remain with the reactor operators while reinforcing in the minds of the public that the United States Government would be accountable in the unlikely event of an accident within United States territory.

8. ***Ports of Entry*** - The alternative ports of entry considered are discussed in Sections 2.2.1.6 and 3.2 of the EIS. The analyses in the EIS demonstrate that the impacts on either the environment, workers or the public due to use of any of the potential ports of entry analyzed would be small and within applicable regulatory limits.

   Although any one or all of the ten ports of entry described in Sections 2.2.1.6 and 3.2 of this EIS would be acceptable ports of entry, DOE and the Department of State concluded that foreign research reactor spent nuclear fuel marine shipments to the United States should be made via the military ports (selected from among those analyzed in the EIS and found acceptable) in close proximity to the spent nuclear fuel management sites. DOE would seek to transport multiple casks per ship to keep the total number of shipments as low as possible, as well as to reduce risks. The exact number of shipments that might be made would be determined by several factors that are unknown at this time, such as the times at which the reactor operators need to make shipments over the 13 year shipping period, the geographic distribution of the reactors, and the availability of suitable ships that would stop at the required ports to pick up and drop off the spent nuclear fuel and target material.

Use of military ports would provide additional confidence in the safety of the shipments due to the increased security associated with the military ports. It could also require much of the spent nuclear fuel to be shipped via chartered ships since commercial ships would not have stops scheduled at military ports, increasing the cost of spent nuclear fuel shipping. This additional cost would be borne by the reactor operators for shipments from high-income economy countries, and by the United States for shipments from other-than-high-income economy countries. Additional costs would be kept to a minimum by shipping as many casks as possible on each ship (up to a maximum of 8 per ship).

9. ***Management Sites*** - The question of which sites should be used for management of all of DOE's spent nuclear fuel was addressed in the Programmatic SNF&INEL Final EIS (DOE, 1995c). That EIS included consideration of the potential receipt of the foreign research reactor spent nuclear fuel. The Record of Decision for that EIS, issued on May 30, 1995, specifies that any aluminum-based foreign research reactor spent nuclear fuel accepted in the United States shall be managed at the Savannah River Site; and that the remaining foreign research reactor spent nuclear fuel shall be managed at the Idaho National Engineering Laboratory. The site for management of the target material was left to be decided under this EIS. All of the target material currently in DOE's possession is managed at the Savannah River Site. The approach incorporated into the preferred alternative is in compliance with the decision specified in the Record of Decision for the Programmatic SNF&INEL Final EIS.

   The analyses in the EIS demonstrate that the impacts to either the environment or the public through use of any of the sites for management of the foreign research reactor spent nuclear fuel and target material would be small, and within the applicable regulatory limits.

10. ***Financing Arrangement*** - The alternative financing arrangements are discussed in Sections 2.2.1.2 and 2.2.2.3 of the EIS. The financing arrangement used for the proposed action would have no effect on the physical processes that would take place, and thus would not have any effect on the potential impacts on the environment, or on the public. However, it could affect how many foreign research reactor operators elect to ship spent nuclear fuel to the United States. For instance, if DOE and the Department of State chose to charge a full cost recovery fee to all reactors, many, if not all, of the reactors in other-than-high-income economy countries would not have the financial resources to participate. On the other hand, if the United States subsidized all of the reactors, the United States would bear the full financial burden, even for reactors which can afford to pay their fair share.

    DOE and the Department of State concluded that, to ensure that reactor operators in other-than-high-income economy countries would participate in the program, the United States should subsidize receipt of their spent nuclear fuel. DOE and the Department of State also concluded that DOE should strive to recover as much of the cost of managing the spent nuclear fuel as possible from high-income economy countries. DOE concluded that it would announce the fee in a *Federal Register* notice, so that the fee may be changed from time to time as necessary to reflect inflation or improvements in DOE's knowledge concerning the costs of the activities to be carried out.

    Such an approach would encourage participation by as many other-than-high-income economy countries as possible, would recover as much as possible of the United States' expenses for management of spent nuclear fuel from high-income economy countries without encouraging any of them to resort to reprocessing of their spent nuclear fuel, and would provide a mechanism through which to account for inflation and future definition of program details.

2.10 Additional Alternatives Considered But Dismissed

Besides ocean transport by vessel, carriage by air is the only other mode of transportation from overseas nations to the United States. There are two distinct reasons why the air mode is not a feasible alternative to the sea mode for transportation of foreign research reactor spent nuclear fuel.

First, with the possible exception of small sample quantities, spent nuclear fuel is required to be transported in packagings (casks) weighing several tons. As a general rule, casks would have to be shipped singly by air (i.e., one per airplane) because of their weight. This has made the air alternative so costly as to be prohibitive. As a result, there is no commercial operational experience in the United States with air transport of spent nuclear fuel. No "Standard Operating Procedures" have been written and no intermodal transfer procedures (air-truck or air-rail) have been developed. No agreements have been negotiated regarding airspace overflight of other nations or states. Because the United States has no experience with this type of transportation, no meaningful comparison can be made between air transport and ship transport regarding either incident-free doses to workers and the public or accident risks.

Second, plutonium air transport packaging standards clearly apply to movement by air of any non-exceptional package containing more than 0.005 curies of plutonium (10 CFR 71.88a). The foreign research reactor spent nuclear fuel considered in this EIS is non-excepted and could contain more than 0.005 curies of plutonium per cask. Therefore, the spent nuclear fuel would have to be transported in a cask meeting plutonium air transport packaging standards. Because no spent nuclear fuel transportation cask has been certified to meet plutonium air transport packaging standards, transporting foreign research reactor spent nuclear fuel by air to the United States could not be accomplished in the near term.

The following additional considerations contributed to the dismissal of air transport as an alternative transportation mode of foreign research reactor spent nuclear fuel: 1) Most United States airports lack rail connections; therefore ground transportation would be limited to the use of trucks; 2) airports have no experience in handling spent nuclear fuel and the capabilities of the available handling equipment are marginal and; 3) worker exposure associated with handling activities would be higher because a lack of automation in handling equipment.

The alternative of accepting of foreign research reactor spent nuclear fuel only from countries that present a potential nuclear weapons proliferation risk was considered but dismissed. A major drawback inherent in this alternative is that potential proliferant countries might well object to being publicly identified as such and, on one pretext or another, refuse to cooperate with the United States in the program. Further, this alternative would not address the potential that some countries that are not currently identified as nuclear weapons proliferation threats might become such a threat in the future. To account for acceptance of foreign research reactor spent nuclear fuel from such countries, DOE would have to assume and analyze one or more "hypothetical reactors" to estimate the potential environmental impacts. The public noted its objection to such an approach when DOE proposed to accept 150 foreign research reactor spent nuclear fuel elements from one or more unnamed "hypothetical reactors" in the first draft of the *Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel* (DOE, 1994m). Finally, implementation of such an alternative would leave unresolved the spent nuclear fuel disposition concerns of the majority of the countries in which foreign research reactors are operating. These countries would be likely to argue, rightly or wrongly, that the United States was not living up to its obligations under the *Treaty on the Non-Proliferation of Nuclear Weapons* to assist nonnuclear weapon states with the peaceful application of nuclear energy. This would damage the credibility of the United States as a reliable partner in the implementation of international nuclear materials management. In consideration of the above summarized flaws, DOE dismissed this alternative from consideration in this EIS.

As a result of public comments, the possibility of managing foreign research reactor spent nuclear fuel on an isolated island was considered, and dismissed. A new facility on an island could not be ready to receive foreign research reactor spent nuclear fuel for at least five years, necessitating temporary management at another location (Savannah River site or Idaho National Engineering Laboratory) for at least the first half of the policy period. Furthermore, management of spent nuclear fuel on such an island is undesirable from the standpoint of security and safety. Provision of physical security would be much more difficult on a remote island than at a mainland site, due to isolation and the greater challenges of protecting open coastlines. Small isolated islands are subject to a greater frequency of severe weather than occurs in the mainland, and after a severe storm it can be more difficult to restore services than it would be in a mainland area.

3. The Affected Environment

This chapter describes the marine, port, and site environments. Marine environments (Section 3.1) would be potentially impacted by the ocean transport of spent nuclear fuel, and port environments (Section 3.2) would be potentially impacted by the transfer of the casks that would contain the foreign research reactor spent nuclear fuel. The affected environment of the potential DOE management sites for storage is addressed in Section 3.3.

3.1 Marine Environment

The ocean is the principal marine environment potentially impacted by foreign research reactor spent nuclear fuel transport. The scientific study of the ocean is commonly referred to as "oceanography." The discipline of oceanography has been subdivided in terms of the basic physical sciences into geological, chemical, physical, and biological oceanography. The purpose of this section is to provide a basic description of the marine environment. It describes those relevant features that have an influence on the general circulation of the world ocean.

3.1.1 Geological Oceanography

Marine geology or geological oceanography is the study of the character and history of that portion of the earth's surface covered by seawater. The world ocean is geographically divided into five major regions: (1) the Southern Ocean, (2) the Atlantic Ocean, (3) the Pacific Ocean, (4) the Indian Ocean, and (5) the Arctic Ocean. The Pacific Ocean occupies roughly 46 percent of the total world ocean area, the Atlantic Ocean approximately 23 percent, the Indian Ocean nearly 20 percent, and the remaining oceans, 11 percent.

The structural features of the ocean basin surface (Figure 3-1) can be divided into five major entities: (1) shore, (2) continental shelf, (3) continental slope and rise, (4) basin (or abyssal plain), and (5) mid-oceanic ridges. The shore region is commonly referred to as that portion of the land mass that has been modified by oceanic processes. The beach is the seaward limit of the shore, and represents a region that is in dynamic equilibrium between the high and low water marks. Extending seaward from the beach face is the continental shelf. It is characterized by a gentle slope of approximately 1:500. The shelf region has an average width of approximately 65 km (40.4 mi), and a water depth of roughly 130 m (426 ft) at the seaward end of the shelf. The continental shelves provide some of the richest fisheries known. At the end of the shelf, the slope drastically steepens (1:20), giving rise to the continental slope, and eventually the continental rise regions. This region averages approximately 4,000 m (13,120 ft) in vertical extent from the shelf to the abyssal plain. The ocean basin constitutes the most extensive area of the ocean bottom surface. Depths in this region range from 3,000 m to 6,000 m (9,840 to 19,680 ft). About 75 percent of the ocean floor is classified as basin area. The deepest areas of the ocean basins are the deep sea trenches, contrasted by the mid-oceanic ridges, which provide relative high points in the ocean bottom surface topography (Pickard and Emery, 1982).

Marine ports are generally located at the confluence of major rivers and the ocean. These regions are commonly referred to as estuaries, and provide a fragile habitat for much of the marine life found in the oceans. An estuary is defined as a semi-enclosed body of water with a free connection to the open ocean,

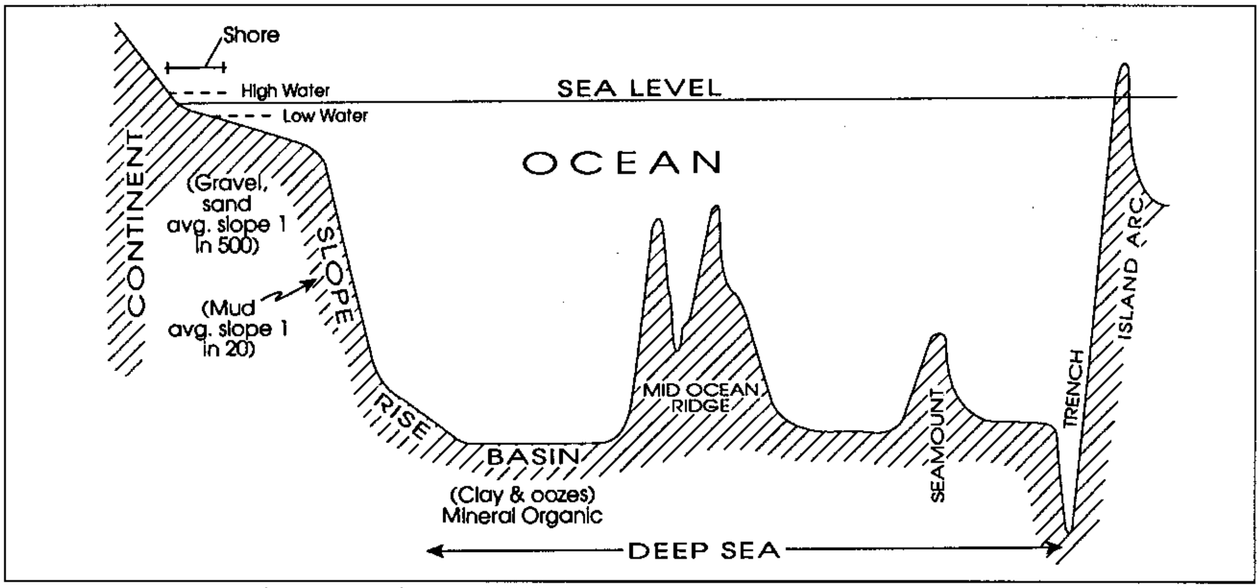


Figure 3-1 Schematic Section Across the Ocean Floor, Depicting Major Geological Features (Pickard and Emery, 1982)

where the saltwater is considerably diluted with freshwater. In general, the freshwater flowing into the estuary eventually exits the system in the upper (water) layer of the estuary, while the denser seawater enters the estuary through lower subsurface layers.

3.1.2 Chemical Oceanography

Seawater is a complex solution of minerals, salts, and elements, containing approximately 80 of the 92 naturally occurring elements. Hydrogen and oxygen, as water, constitute the largest elemental percent of seawater, with sodium chloride (NaCl) being the most abundant salt (78 percent) in the solution. Magnesium, calcium, and potassium chlorides and carbonates provide the bulk of the remaining constituents of the seawater solution. The ratio of these elements within the solution is relatively constant from ocean to ocean. However, in coastal areas where freshwater river influences are significant, the water chemistry can be substantially different. In addition to the major and minor constituents described above, trace metals, nutrient elements, dissolved atmospheric gases, and other organic matter also form important components of seawater. While trace metals are essential to the growth and development of certain organisms at low concentrations, these elements can become toxic when concentrated at high levels. Table 3-1 summarizes the concentration of major elements and trace elements, expressed in milligram per liter (mg/L), in seawater. The major nutrients (phosphates, silicates, and nitrates) provide the chief limiting agent for oceanic phytoplankton production. Atmospheric gases (e.g., oxygen and carbon dioxide) absorbed by the ocean play important roles in the overall global climate of the earth.

Naturally occurring radionuclides of uranium (such as ^{234}U, ^{235}U, ^{238}U), and polonium-210 (^{210}Po), are present in seawater, and in marine organisms, at concentrations generally greater than those found in terrestrial ecosystems. The ocean water concentrations of uranium isotopes are: ^{234}U, 1.30 picocuries per liter (pCi/l); ^{235}U, 0.05 pCi/l; and ^{238}U, 1.2 pCi/l (IAEA, 1976). For comparison, other major radioisotopes found in ocean water are: potassium-40 (40K), 486 pCi/l; thorium-232 (232Th), 540 pCi/; tritium (3H), 3 pCi/l; rubidium-87 (87Rb), 3 pCi/l; and Carbon-14 (14C), 1.8 pCi/l (IAEA, 1988).

Table 3-1 Concentration of Major Elements and Trace Elements in Seawater (CRC, 1991)

Element	mg/L	Trace Element	mg/L
Chlorine	19,000	Strontium	8.1
Sodium	10,500	Arsenic	0.003
Magnesium	1,350	Iron	0.01
Sulphur	885	Copper	0.003
Calcium	400	Zinc	0.01
Potassium	380	Cesium	0.0005
Bromine	65	Uranium	0.003
Fluorine	1.3	Lead	0.00003
Iodine	0.06	Zirconium	0.000022

The relationship between environmental concentrations of radionuclides and the concentration found in organisms is important in the study of food chain effects. Bioamplification, the increase in concentration of radionuclides in organisms progressively further up the food chain (as with organic pesticides in terrestrial environments), is observed in marine food chains. In the marine environment, uranium has not been found to bioamplify in fish, and there is only slight bioamplification in crustaceans and mollusks (IAEA, 1976). The readiness with which other radionuclide constituents of spent nuclear fuel may enter the food chain is variable, but generally low.

3.1.3 Physical Oceanography

The science of physical oceanography involves the development of a systematic quantitative description of ocean characteristics and circulations. Ocean circulations include not only the major, permanent ocean features (e.g., the Gulf Stream) that circulate continuously with fluctuating velocity and position dynamics, but also the smaller-scale circulation features (e.g., tides, waves, coastal currents, etc.). Gradients in temperature, salinity, and seawater density give rise to vertical and lateral circulations.

The primary forces behind the generation and maintenance of surface currents in the world ocean are the winds in the lower portions of the atmosphere. Low-level winds generate stresses on the ocean surface that give rise to the surface currents. However, these currents only affect the uppermost layers of the ocean. Thus, the global wind patterns establish the direction and magnitude of the surface currents. Figure 3-2 depicts the major components of the wind-induced surface circulation of the world ocean.

Northern hemisphere ocean basins are characterized by strong western basin boundary currents that transport warmer, less dense water poleward, and are balanced by weaker, colder return flows along the eastern basin boundaries. Examples of these flows in the northern hemisphere are the Gulf Stream and Kuroshio currents, and the Canary and Californian currents, respectively. These permanent circulation features are the result of the strong mid-latitude westerly winds and the easterlies in the tropics. Due to the strength of these oceanic and atmospheric circulations, North Atlantic and North Pacific shipping routes tend to follow these flows.

Also of interest are the deep water convective circulations, which are linked with the surface system circulation. In general, these circulations are generated in high latitudes by air-sea interaction processes producing relatively cold and dense surface waters that sink and flow into the central ocean basins. This loss of water in the high latitudes is replaced by warmer surface waters migrating poleward at intermediate depths. Thus, in considering the overall environmental impact of the proposed and alternative actions, the intermediate and bottom water masses/circulations cannot be ignored, due to their surface origin.

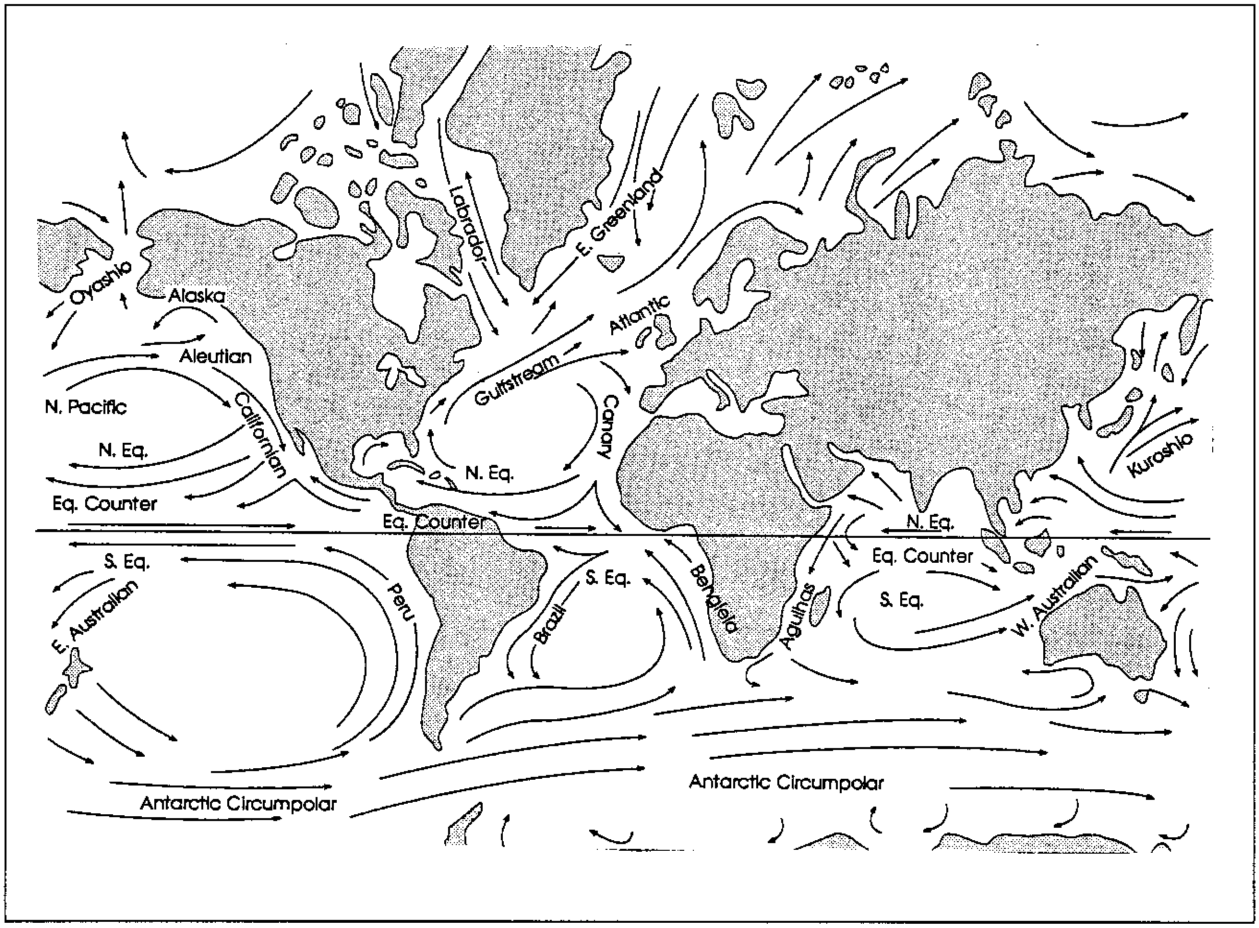


Figure 3-2 Major Wind-Driven Surface Currents of the World Ocean (Kennett, 1982)

3.1.4 Biological Oceanography

Biologically, the characteristics of ocean organisms dramatically change with ocean depth. Changes in organisms can be correlated with the decrease in the amount of light and the wavelength of the light that penetrate to a given depth. This variation in light is also influenced by the turbidity of the oceanic waters, and has a great influence on the biological productivity of a given region. Upper water layers are rich in nutrients and more productive than water layers found at depths greater than 200 m (660 ft). Abundant plant life supports the many animal species found at depths less than 200 m (660 ft). The estuarine areas found at the margins of the shelf region and the continents provide rich, productive breeding and spawning grounds for many marine organisms. In contrast, the deep ocean bottoms are limited in productivity because of the absence of light and the scarcity of nutrients (Friedrich, 1969).

The deep sea bottom dwellers are highly diverse, with many biological groups represented by more species than in most shallow-water communities (Hessler and Sanders, 1967). However, the number of individual organisms in a given volume does decrease in the deep sea and this, together with a general tendency toward decrease in the average size of the organisms, results in a dramatic reduction in standing stock or biomass on the deep ocean floor. In round figures, the total wet weight of bottom-living organisms in and on each square meter (m) of seabed decreases from 10-100 grams (g) on the continental shelf, to 1-10 g on the continental slope, and to only 0.1-1.0 g on the abyssal plain (Rice, 1978).

3.2 Individual Port Marine Environments

This section presents general environmental information for ten U.S. ports that have been identified as potential ports of entry. The ten ports are:

> Charleston, SC [includes the Naval Weapons Station (NWS) at Charleston and the Wando Terminal]; Galveston, TX; Hampton Roads (includes terminals at Newport News, Norfolk, and Portsmouth), VA; Jacksonville, FL; the Military Ocean Terminal at Sunny Point (MOTSU), NC; the NWS at Concord, CA; Portland, OR; Savannah, GA; Tacoma, WA; and Wilmington, NC.

These ports are more fully described in Appendix D of this Environmental Impact Statement (EIS). Appendix D identifies the ports that were considered as potential ports of entry for foreign research reactor spent nuclear fuel, the criteria used in the port evaluation process, the method of evaluation, and the results of the evaluation process. Appendix D also presents population data for ports and transportation routes considered in the evaluation. Potential overland and barge transportation routes are described in Appendix E, and Appendix C presents information on the environmental impacts of marine transport.

The various policy, management, and implementation alternatives being considered in this EIS do not involve any construction or modification of port facilities, nor would the use of one or more ports for the receipt of foreign research reactor spent nuclear fuel be expected to noticeably increase the number of vessel calls to the port or interfere with existing port operations. Once at the port of destination, the spent nuclear fuel would be transferred from the vessel to a waiting truck or train and shipped to the destination as expeditiously as possible.

3.2.1 Environmental Information for the Potential Ports of Entry

This section presents summary environmental information for the potential ports of entry for foreign research reactor spent nuclear fuel.

3.2.1.1 Charleston, SC (Includes Terminals at the Naval Weapons Station and the Wando Terminal)

Charleston is the largest port city in South Carolina, and the greater Charleston area is one of the major seaports on the East Coast of the United States. The city of Charleston is located at the confluence of the Cooper and Ashley Rivers, approximately 11 km (7 mi) west of the Atlantic Ocean. The principal wharves are along the west bank of the Cooper River, except for the Wando Terminal, which is along the east bank of the Wando River near the city of Mount Pleasant, about 20 km (12 mi) from the Atlantic Ocean. The city of Charleston is on a peninsula, bounded on the west and south by the Ashley River and on the east by the Cooper River. In general, the elevation of the area ranges from sea level to approximately 6 m (20 ft) on the peninsula.

Environmental Conditions: The State of South Carolina has classified the water quality of the lower portion of the Wando River as both SFH and SA (SFH waters are shellfish harvesting waters, and SA waters are suitable for primary and secondary recreation and for other water uses requiring lower water quality). According to the U.S. Fish and Wildlife Service's Ecological Inventory Map for James Island, SC, the Wando Terminal and the NWS Charleston are located in a mid-salinity estuarine habitat (generally 5 to 16.5 ppt). The Charleston harbor which is traversed enroute to either terminal, is located in a high-salinity estuarine habitat (generally 16.5 to 30 ppt) (FWS, 1980a).

The State of South Carolina has classified the water quality of the portion of the Cooper River above the confluence with the Ashley River as SB (SB waters are tidal saltwaters suitable for secondary contact recreation, crabbing, and fishing, except the harvesting of clams, mussels, or oysters for market purposes and human consumption). The waters of Goose Creek, upstream of the confluence with the Cooper River to the dam at the Charleston Waterworks, are also Class SB (Department of the Navy, 1994).

State or Federally protected endangered or threatened aquatic species in the vicinity of the Charleston harbor include the shortnose sturgeon, Atlantic sturgeon, and the American shad. Bachman's warbler is a Federally protected bird species also found in the vicinity (FWS, 1980a). While there are some wetlands in the vicinity of Wando Terminal and on the property of NWS Charleston (Department of the Navy, 1990 and 1994), there are no known special wildlife sanctuaries or habitats of concern in the general area. Bald eagles have been observed on the NWS Charleston property and are believed to be nesting in the far northern areas of the Station. Red-cockaded woodpeckers are known to inhabit NWS Charleston. Although, the hurricane Hugo (September 1989) destroyed much of their habitat (mature pine trees with red heart disease), several colonies are surviving with the assistance of artificial nest bates (Lewis, 1995). The Charleston harbor area and the west bank of the Cooper River are commercially well developed.

The lower Wando and Cooper Rivers and the Charleston harbor support a large number of aquatic and terrestrial species. Aquatic species commonly found in the vicinity include crabs, oysters, clams, shrimp, sturgeon, herring, shad, seabass, kingfish, drum, flounder, and mackerel. Marine mammals, including dolphins and whales, have been sighted in the harbor. According to the South Carolina Heritage Trust, no rare, threatened, or endangered species or communities have been recorded in the area near the Wando Terminal (McBee, 1994).

Climatic Conditions: The climate of this region is temperate, primarily due to its close proximity to the Atlantic Ocean. The prevailing winds are generally northerly in the fall and winter months, becoming more southerly during the summer months. This type of circulation serves to "warm" the region during winter and "cool" it during the summer. Summer is the rainy season in Charleston, with the city receiving 41 percent of the annual total rainfall during the summer months. Except for the occasional tropical storm or hurricane, the majority of this rain occurs during afternoon and evening thunderstorms. The late summer and early fall brings the highest probability of tropical storm activity to the Charleston area. The fall season is a transitional period, where temperature extremes are rare and sunshine is abundant. The winters in this area are mild with periods of rain. However, in contrast to the summer, winter rains tend to be steady and uniform, and last for several days. The most unstable period in this region is spring, when the confluence of warm moist tropical air and cool dry continental air increase the occurrence of severe weather in this region. The average earliest freeze is in early December, and the average last frost is in late February (NOAA, 1992c).

The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Charleston, the Uniform Building Code provides a basic wind speed of about 160 km per hour (100 m per hour). The greater Charleston area is located in a moderate seismic zone with an acceleration of 0.15 g. The effective peak velocity-related acceleration represents the back-and-forth horizontal motion of the ground due to a seismic event at a period of 1.0 sec. This acceleration is expressed in relation to g, where g equals acceleration due to gravity.

Naval Weapons Station - Charleston: The NWS is located on the west bank of the Cooper River, north of the city of North Charleston. The NWS is approximately 7080 hectares (17,500 acres) in size and is located in southeastern Berkely County, South Carolina, about 30 km (19 mi) from the Atlantic Ocean. The NWS has two useful wharves and two useful piers. Wharf Alpha and Pier Bravo have cranes and are capable of loading trucks or trains directly from the ships. Pier Charlie and the Military Traffic Management Command Terminal would have to use shipboard or mobile cranes to load trucks. Several facilities on the NWS could be used to transfer spent fuel casks or containers from trucks to rail cars. A map of the port is shown in Figure 3-3.

The 1990 population within 16 km (10 mi) of the Wharf Alpha was 209,188. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five U.S. Department of Energy (DOE) management sites are: the Savannah River Site, 46,200; the Oak Ridge Reservation, 108,000; the Idaho National Engineering Laboratory, 498,000; the Hanford Site, 550,000; and the Nevada Test Site, 540,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 303 km (188 mi), the Oak Ridge Reservation, 647 km (402 mi), the Idaho National Engineering Laboratory, 3,930 km (2,442 mi), the Hanford Site, 4,601 km (2,859 mi), and the Nevada Test Site, 4,094 km (2,544 mi). Distances along rail routes are slightly longer.

Ethnic and Income Characteristics: Figure 3-4 shows the ethnic composition for the area surrounding the port at the NWS Charleston. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans constituted about 31 percent of the total population, and approximately 88 percent of the minority population for the area surrounding the port. Figure 3-5 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

Wando Terminal: This South Carolina State Port Authority terminal is located at the confluence of the Wando and Cooper rivers, on the east bank of the Wando River, near the incorporated city of Mount Pleasant. The facility has three modern container berths, with a fourth under construction, and a large paved container storage yard. The Wando terminal is about 8.1 km (5 mi) from the nearest Interstate highway and 15 km (9 mi) from the nearest intermodal rail yard. A map of the port is shown in Figure 3-6.

The 1990 population within 16 km (10 mi) of the Wando Terminal was 233,424. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five U.S. Department of Energy (DOE) management sites are: the Savannah River Site, 65,700; the Oak Ridge Reservation, 127,000; the Idaho National Engineering Laboratory, 518,000; the Hanford Site, 569,000; and the Nevada Test Site, 559,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 327 km (203 mi), the Oak Ridge Reservation, 671 km (417 mi), the Idaho National Engineering Laboratory, 3,954 km (2,457 mi), the Hanford Site, 4,625 km (2,879 mi), and the Nevada Test Site, 4,118 km (2,559 mi). Distances along rail routes are slightly longer.

Ethnic and Income Characteristics: Figure 3-7 shows the ethnic composition for the area surrounding the Wando Terminal. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans constituted about 33 percent of the total population, and approximately 93 percent of the minority population for the area surrounding the port. Figure 3-8 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

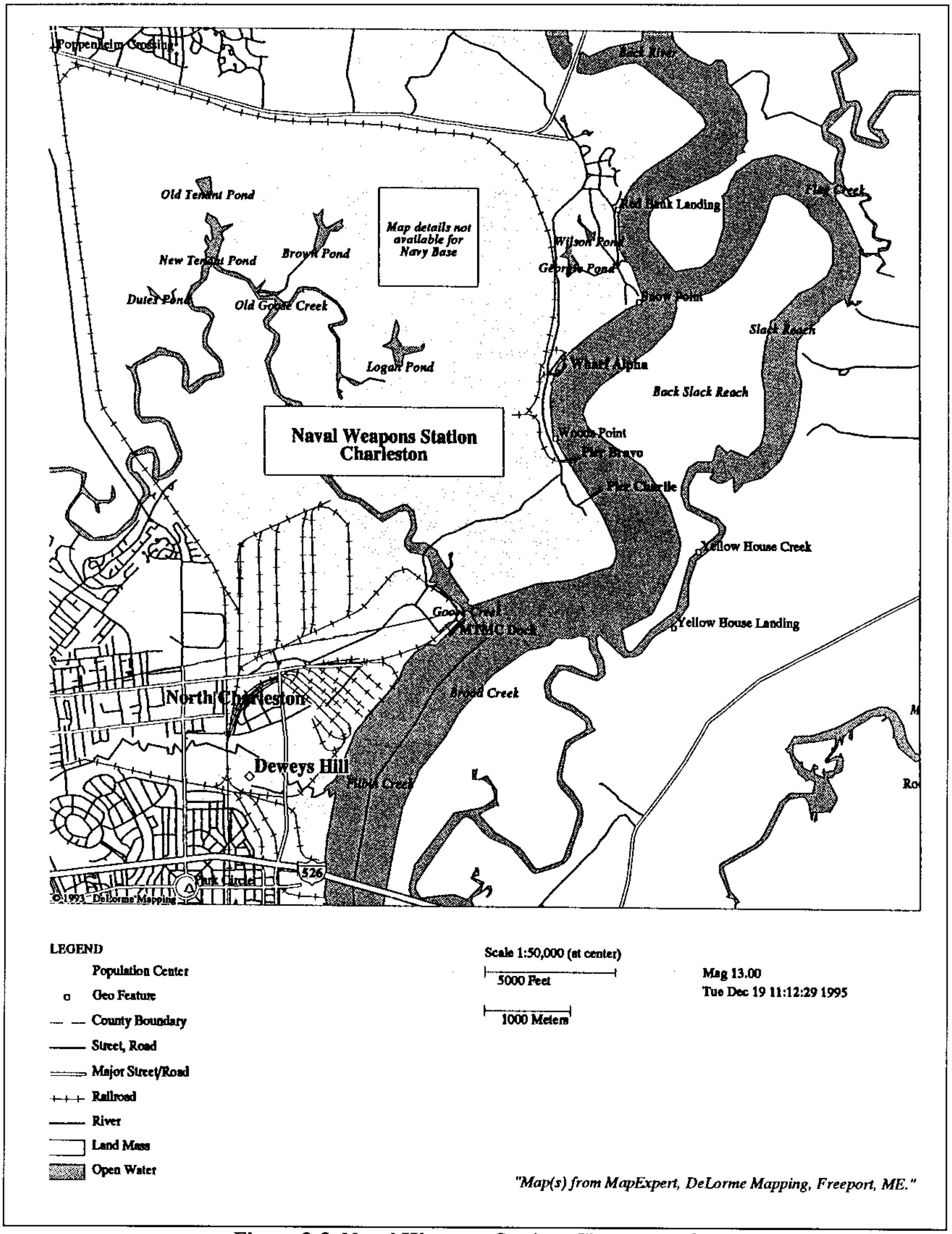


Figure 3-3 Naval Weapons Station, Charleston, SC

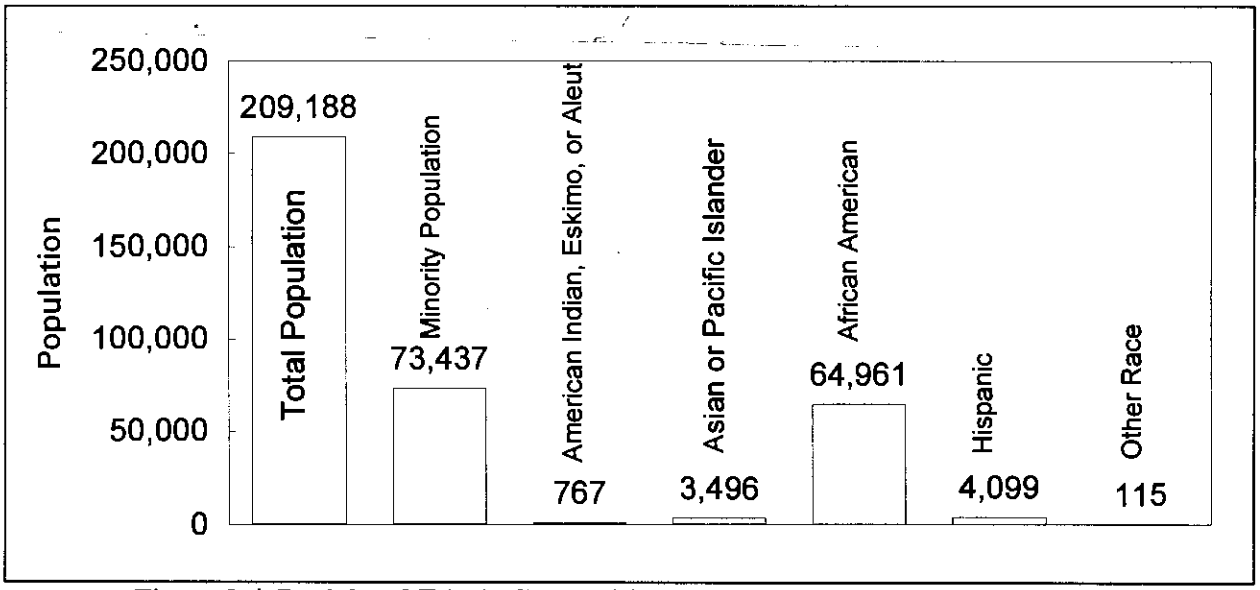


Figure 3-4 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Naval Weapons Station, Charleston

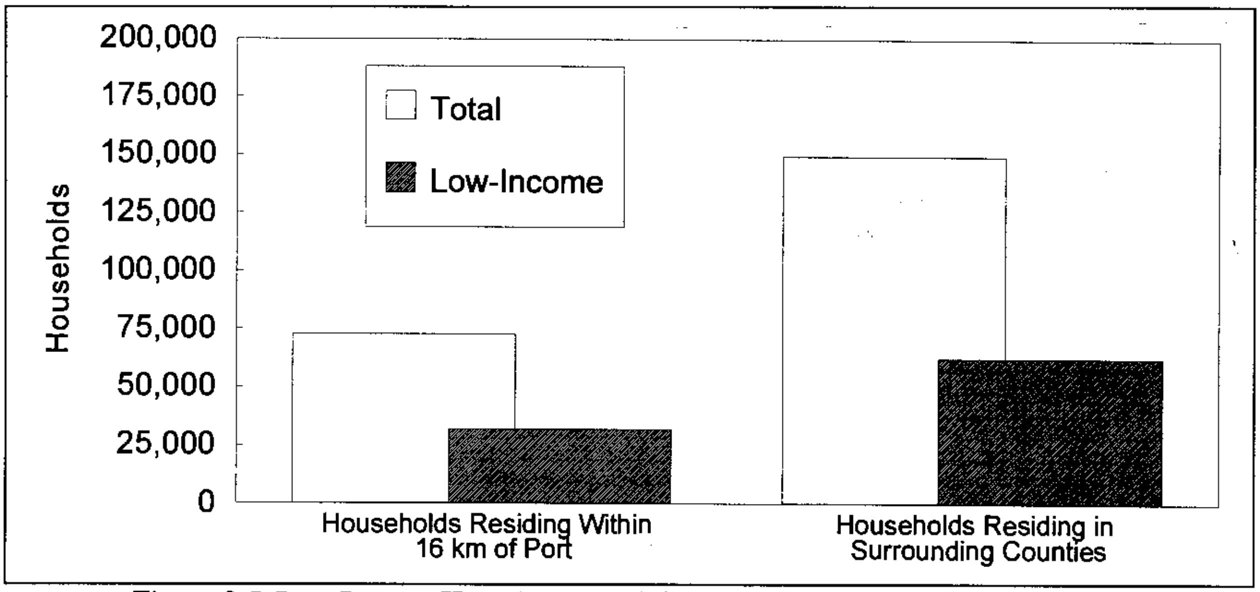


Figure 3-5 Low-Income Households Residing within 16 km (10 mi) of the Naval Weapons Station, Charleston

3.2.1.2 Galveston, TX

Galveston, TX is situated within 16 km (10 mi) of the entrance to the Gulf of Mexico. The city of Galveston occupies the entire width of the east end of Galveston Island. The shipping wharves are on the north side of the island and the Gulf of Mexico is on the south. The Port of Galveston is located in the heart of the city. A map of the port is shown in Figure 3-9.

Galveston is a major resort and tourist center for the Southwest. There is a waterfront tourist attraction at "Pier 21" close to the port area. A public park on Pelican Island, reached by causeway, is located across the Intracoastal Waterway from the Port of Galveston. A cruise ship terminal is located at Pier 25 in the

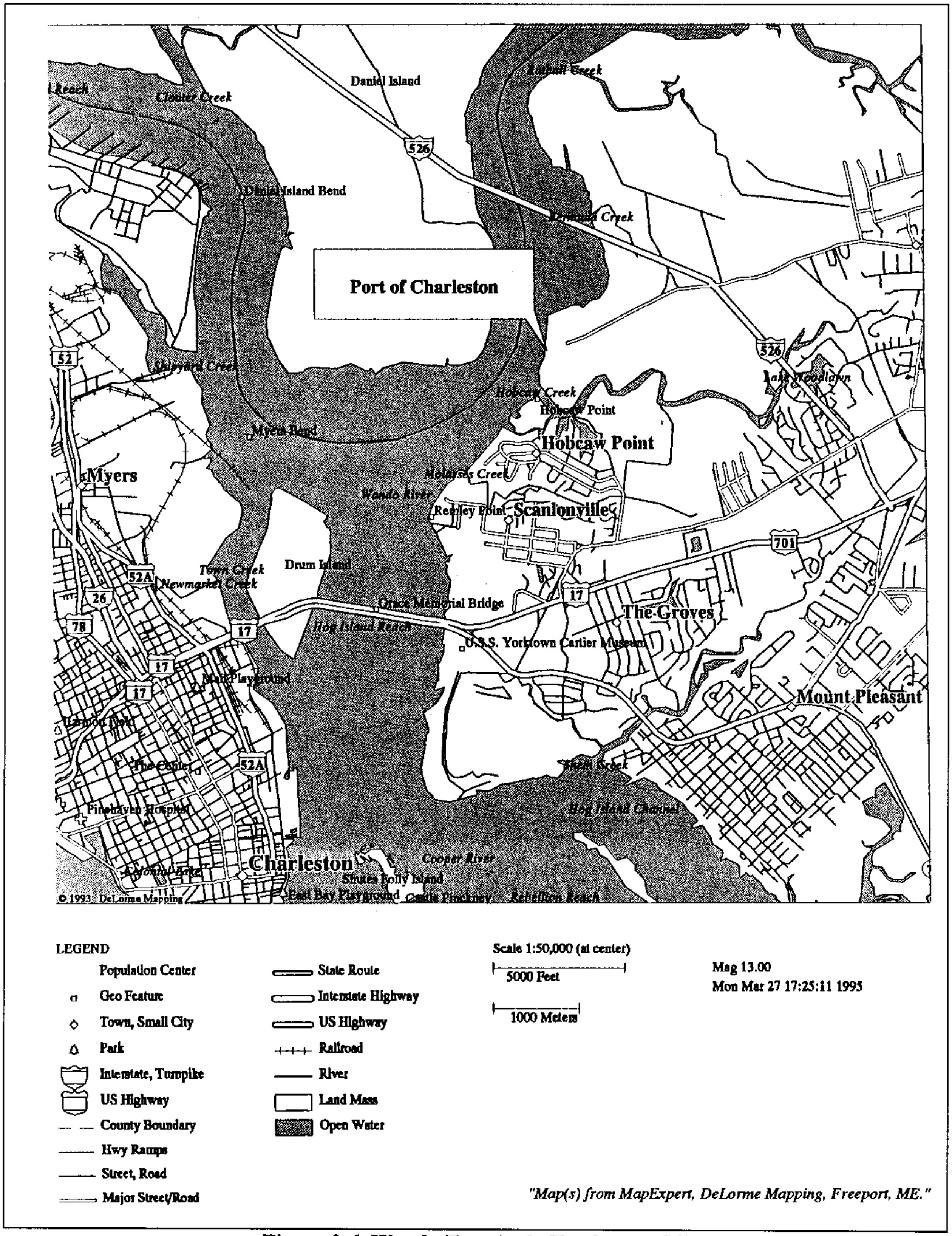


Figure 3-6 Wando Terminal, Charleston, SC

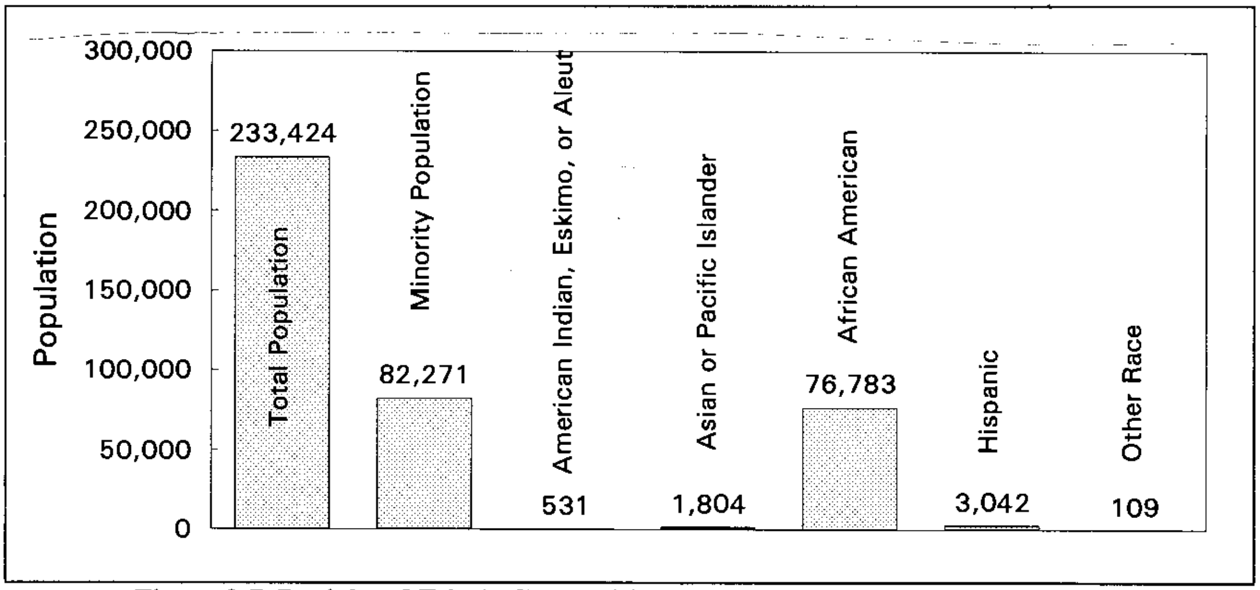


Figure 3-7 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Wando Terminal, Charleston

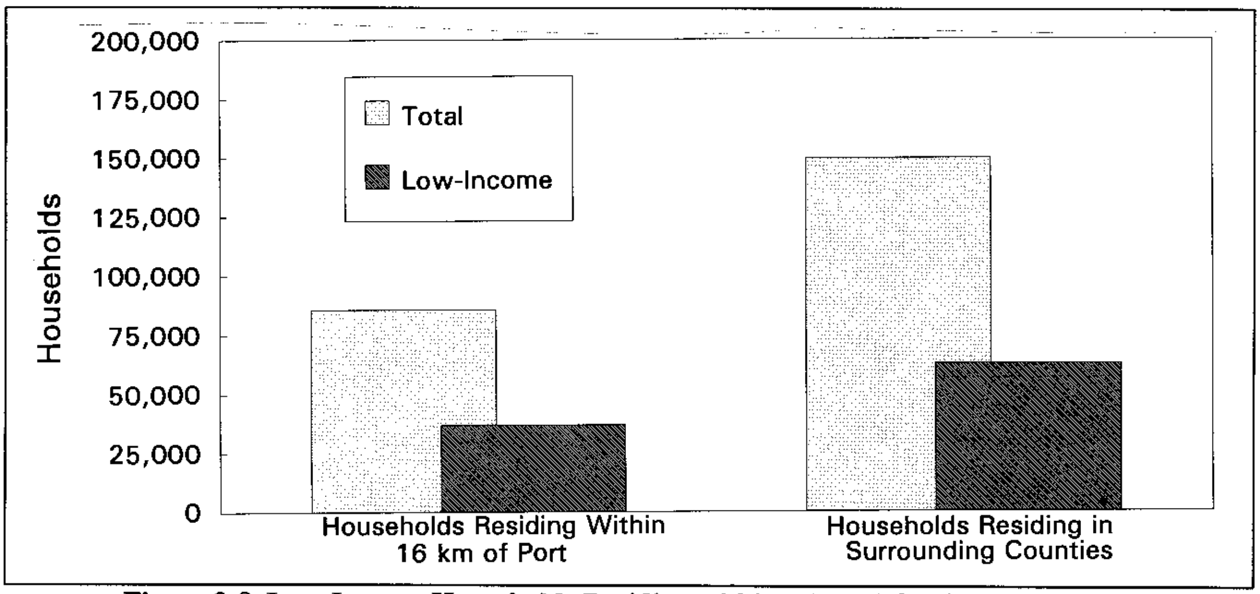


Figure 3-8 Low-Income Households Residing within 16 km (10 mi) of the Wando Terminal, Charleston

heart of the Port of Galveston complex, and there is a tanker terminal on Pelican Island across from the Port of Galveston at its southern end. A Federal project provides for an entrance channel, and an outer bar channel both dredged to 12.8 m (42 ft).

The Port of Galveston's principal container handling facility is the container terminal at Pier 10. This facility has a controlled all-weather truck entrance and interchange area. The terminal is connected to Interstate Highway 45 on the mainland by a 9.3 km (5.8 mi) four-lane State highway and two 2.8 km (1.75 mi) causeways that cross the southwest end of Galveston Bay. The container handling facility is served by four major railroads, the Burlington Northern, Santa Fe, Southern Pacific, and Union Pacific

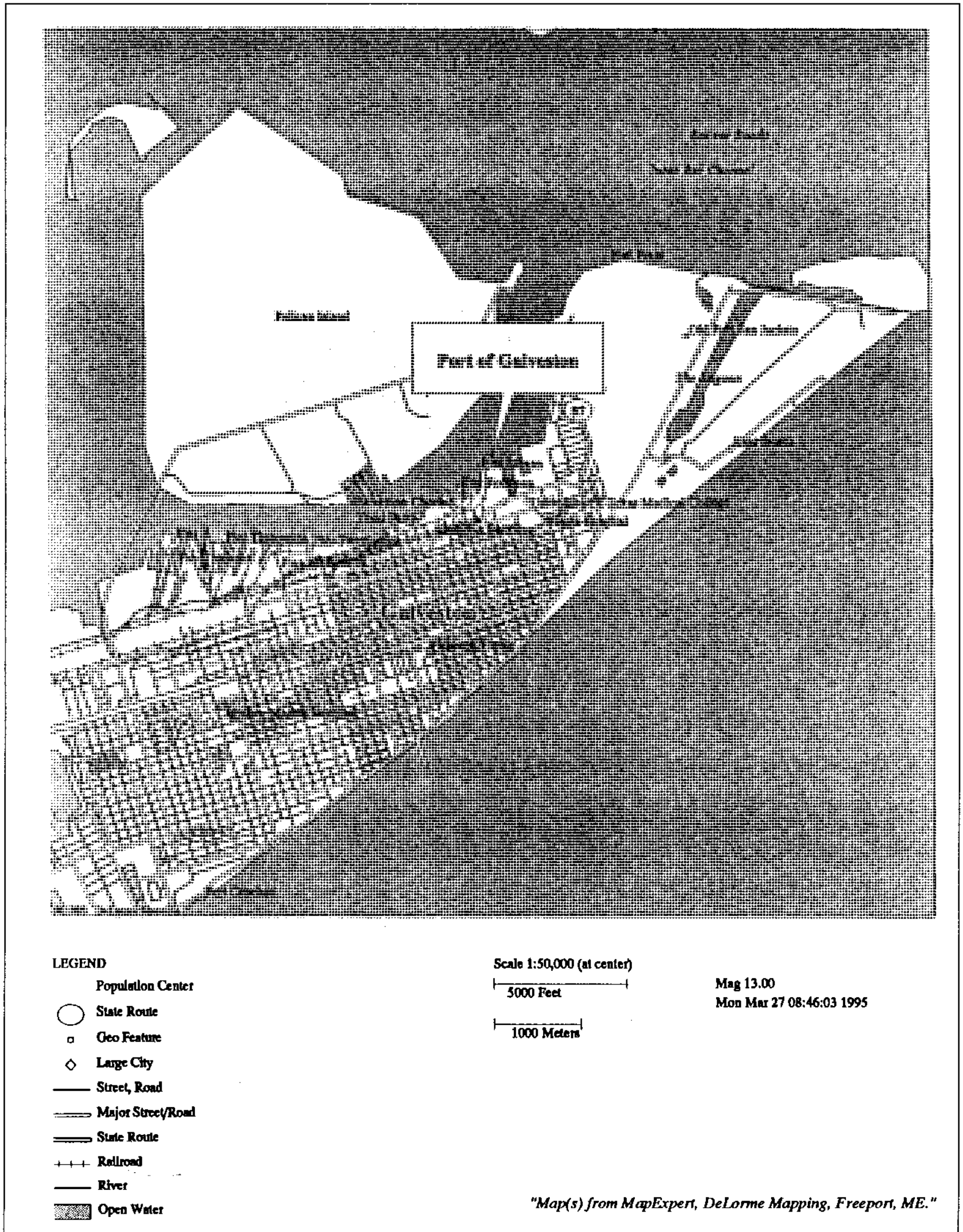


Figure 3-9 Port of Galveston, TX

Lines. Galveston Railway, Inc., provides terminal connections and performs switching of all port rail traffic. An intermodal container transfer terminal is located within the container terminal, and trackage extends to within 30.5 m (100 ft) of ship berths.

The 1990 census population within 16 km (10 mi) of the port terminals was 73,322. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 403,000; the Oak Ridge Reservation, 337,000; the Idaho National Engineering Laboratory, 526,000; the Hanford Site, 575,000; and the Nevada Test Site, 595,000. Populations along rail routes to these sites are slightly larger for the Savannah River Site and the Oak Ridge Reservation, but slightly less for the Idaho National Engineering Laboratory, the Hanford Site, and the Nevada Test Site. The distances to the five potential sites on interstate routes are: the Savannah River Site, 1,600 km (1,000 mi); the Oak Ridge Reservation, 1,550 km (963 mi), the Idaho National Engineering Laboratory, 3,070 km (1,908 mi); the Hanford Site, 3,740 km (2,324 mi); and the Nevada Test Site, 3,000 km (1,864 mi). Distances along rail routes are slightly longer.

Environmental Conditions: A large number of aquatic and terrestrial species frequent the Galveston Bay area. A variety of birds migrate, winter, and breed along the Texas Coast including shorebirds, songbirds, waterfowl and raptors (Breslin, 1993; FWS, 1992). These endangered/threatened bird species include the brown pelican, peregrine falcon, bald eagle, Atwater's greater prairie-chicken, piping plover, and the Eskimo curlew (State-threatened only). Endangered/threatened marine mammals and sea turtles also are found in the coastal bay systems and the Gulf of Mexico. Galveston Bay is within the range of the green, hawksbill, Kemp's ridley, leatherback, and loggerhead sea turtles. While no protected species are known to be located within the Port of Galveston, significant populations of the endangered brown pelican and the threatened piping plover exist nearby (Werner, 1994). The U.S. Fish and Wildlife Service reported that as many as 600 brown pelicans have been sighted loafing on the north end of Little Pelican Island, which is approximately 5.6 km (3.5 mi) northwest of the port. In addition, approximately 125 pairs nested and produced 90 young ones at this site in 1994. This was the first time that brown pelicans had successfully nested in Galveston Bay in over 40 years. Wintering populations of the threatened piping plover use the northeastern end of Galveston Island and the southeastern end of Bolivar Peninsula. Of the 3,187 birds observed during the first Gulf Coast count of wintering piping plovers, 1,904 were observed on the Texas coastline (Werner, 1994).

A great amount of commercial and recreational fishing occurs in Galveston Bay and the Gulf of Mexico. Shellfish are the most important commercial species, particularly shrimp followed by eastern oysters and blue crabs (TPWD, 1989a). The most valuable finfish landed from the Galveston Bay system are black drum and mullet. In 1988, a total of 507,7169 kg (11,169,773 lb) of shellfish valued at $13,489,146 was landed from the Galveston Bay System; a total of 224,536 kg (493,980 lb) of finfish valued at $226,140 was also landed. The major recreational species of fish that were caught in the Galveston Bay system in 1987-1988 were: Atlantic croaker, sand seatrout, spotted seatrout, southern flounder, black drum, and red drum (TPWD, 1989b). Galveston Bay has been named as an "estuary of national significance" by the U.S. Congress. The implementation of the proposed action would pose no significant radiological or non-radiological risks to the environment in the Galveston area, including estuaries.

Climatic Conditions: The climate of the Galveston area is predominantly marine, with periods of modified continental influence during winter. The port is subject to hurricanes and tropical storms (NOAA, 1993a). The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. For the Port of Galveston, the Uniform Building Code provides a basic wind speed of about 160 km per hour (100 mph) (UBC, 1991). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

Ethnic and Income Characteristics: Figure 3-10 shows the ethnic composition for the area surrounding the Port of Galveston. This figure shows the population residing within 16 km (10 mi) of the port, according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 27 percent of the total population, and approximately 54 percent of the minority population for the area surrounding the port. Hispanics made up about 20 percent of the total population, and approximately 40 percent of the minority population around the port. Figure 3-11 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

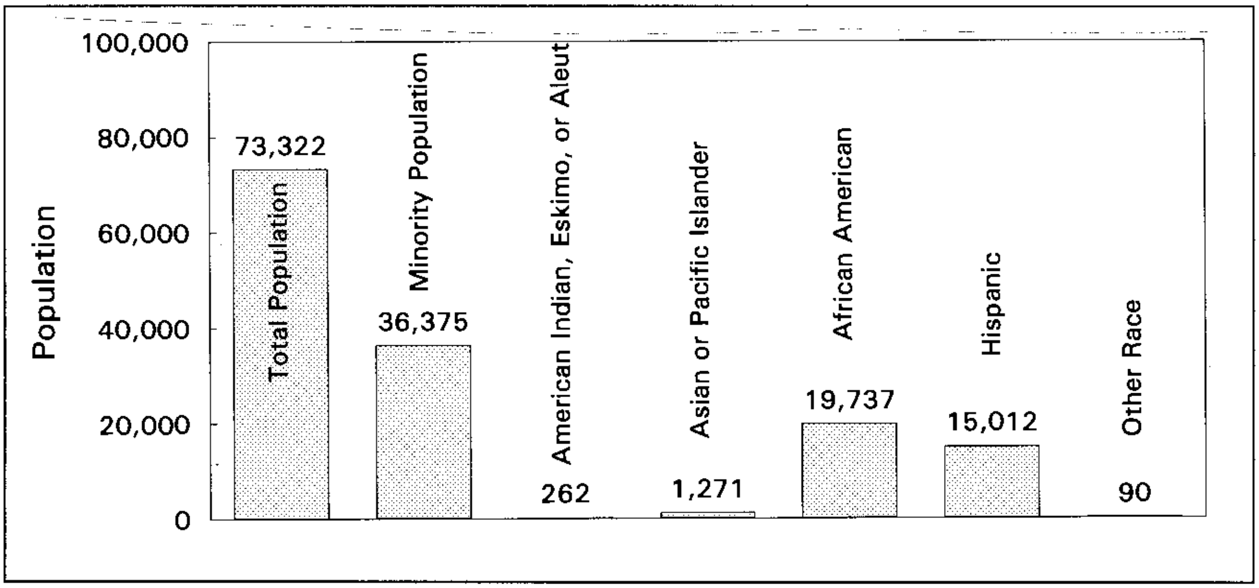


Figure 3-10 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Galveston

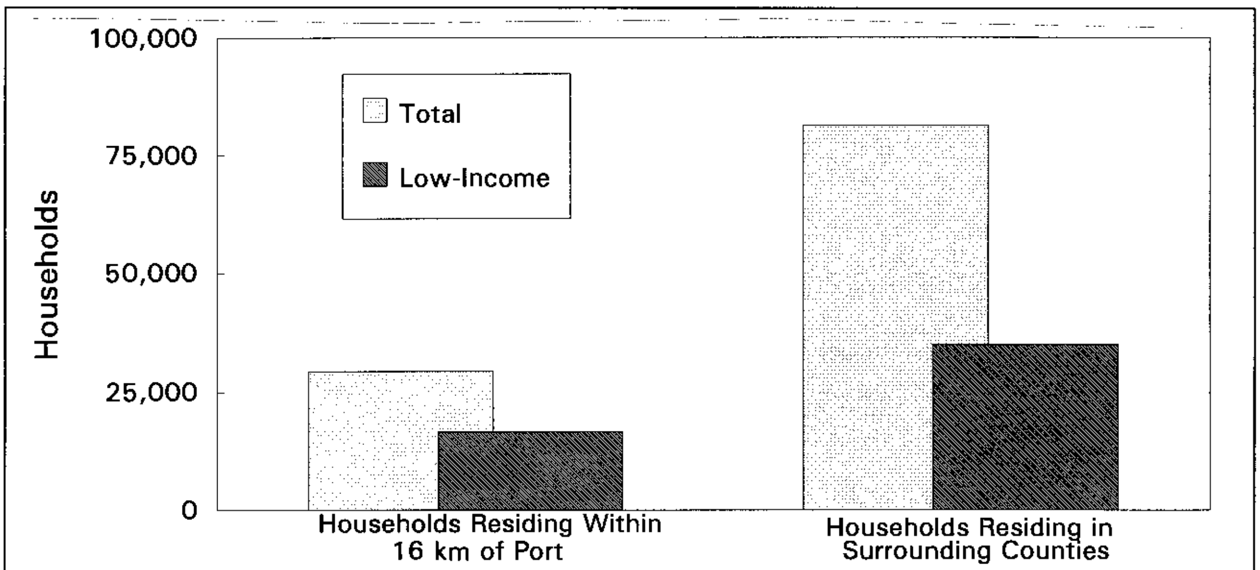


Figure 3-11 Low-Income Households Residing within 16 km (10 mi) of the Port of Galveston

3.2.1.3 Hampton Roads, VA (Includes Terminals at Newport News, VA; Norfolk, VA; and Portsmouth, VA)

Hampton Roads is one of the world's foremost bulk cargo harbors, and has more collective experience handling spent nuclear fuel than any other port in the United States. It is a multi-terminal port with privately and publicly owned marine cargo handling facilities, and is located at the southwest corner of the Chesapeake Bay at the confluence of the James and the Elizabeth Rivers. The port is about 26 km (16 mi) from the Virginia Capes and the entrance from the Atlantic Ocean. The major terminals located on the Elizabeth and James Rivers are approximately another 10 to 13 km (6 to 8 mi) from the Chesapeake Bay. The port includes the port terminals at Norfolk, Portsmouth, and Newport News. All three terminals are located in commercial port districts of their respective cities, somewhat separated from other community activities, in areas dedicated primarily to port industrial usage. Adjacent communities include the cities of Chesapeake and Virginia Beach.

Environmental Conditions: The Port of Hampton Roads is located at the confluence of the James River and the Chesapeake Bay, approximately 29 km (18 mi) west of the Atlantic Ocean. The average elevation of this region is approximately 4 m (13 ft) above sea level. There are no known areas of special environmental concern other than the growing interest in preservation of the Chesapeake Bay and its tributary rivers. The Dismal Swamp National Wildlife Refuge is located about 16 km (10 mi) from the two terminals on the Elizabeth River, but water drainage from the swamps is toward the port area. The swamp refuge is far enough from the terminals that potential negative impacts of low-probability, severe accidents in the ports on wildlife populations would be negligible. The three port terminals at Hampton Roads are described separately below.

Climatic Conditions: The geographic location of this region is especially favorable, tending to be located south of the predominant winter extratropical cyclone tracks which originate at higher latitudes and north of the usual tropical cyclone (e.g., tropical storms and hurricanes) paths. In general, the winters are mild with slightly warmer temperatures during the spring and fall seasons. The summer season is warm and long, but is characterized by frequent cool periods, generated by cool northeasterly winds off of the North Atlantic. Extreme cold waves are infrequent, and temperatures below -18°C (0°F) are almost nonexistent. In general, winters pass without measurable snowfall and most snowfall melts within 24 hours. The average first sub-freezing day in the fall is November 17th and the last occurrence in the spring is March 23rd. The predominant wind directions since 1984 are from the south-southwest and north-northeast and vary seasonally (NOAA, 1992c).

The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Hampton Roads, the Uniform Building Code provides a basic wind speed of about 140 km per hour (90 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

Newport News Marine Terminal: This terminal is located on the north shore of the Port of Hampton Roads on the James River. It is a combination container, roll-on/roll-off, and breakbulk terminal. The facility has two piers, two container vessel berths, and four container cranes. There is covered storage on both piers. A map of the Newport News Marine Terminal is shown in Figure 3-12.

The 1990 population within 16 km (10 mi) of the port terminal was 430,757. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 181,000; the Oak Ridge Reservation, 209,000; the Idaho National Engineering Laboratory, 628,000; the Hanford Site, 677,000; and the Nevada Test Site, 691,000. Populations along rail

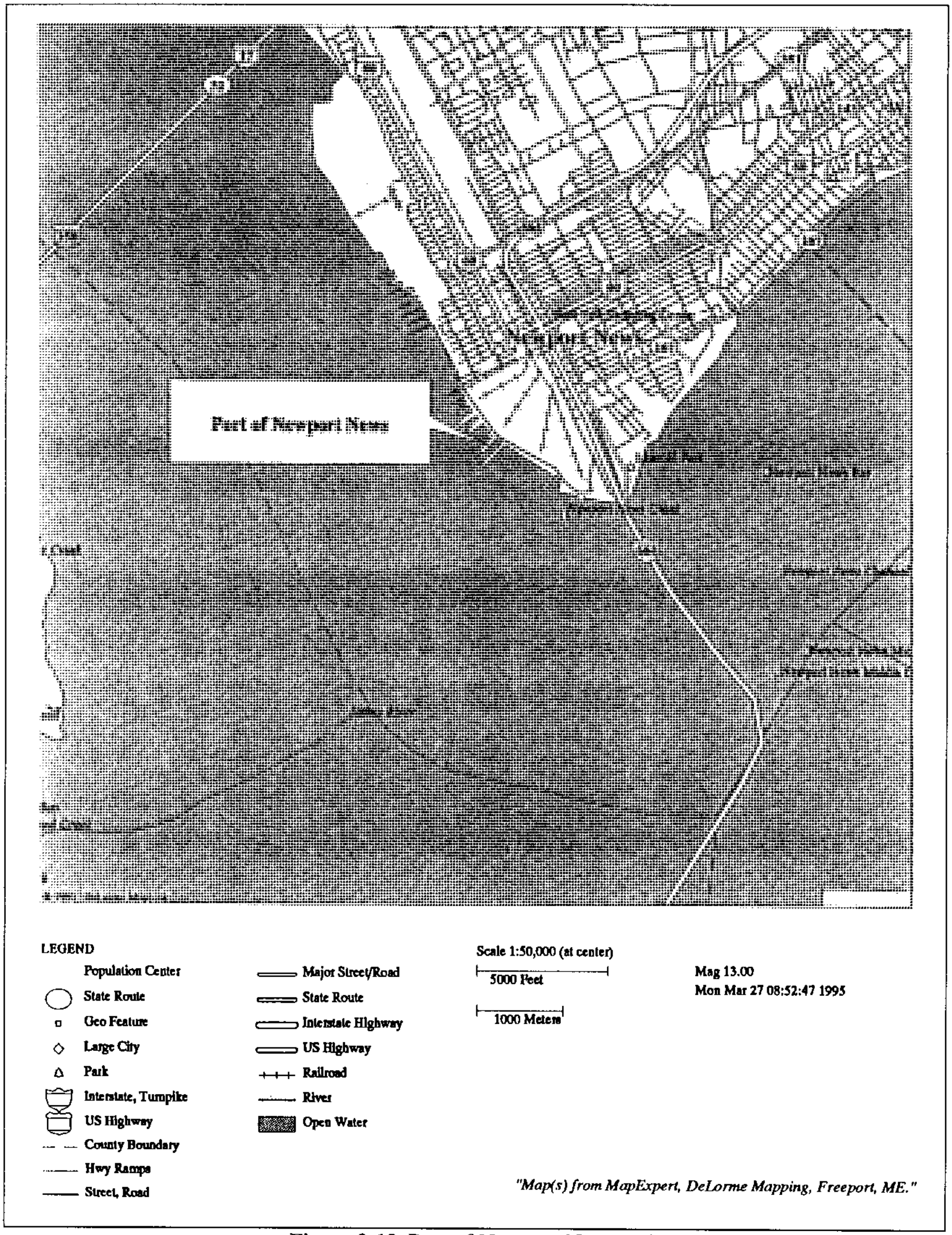


Figure 3-12 Port of Newport News, VA

routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 840 km (522 mi); the Oak Ridge Reservation, 890 km (553 mi); the Idaho National Engineering Laboratory, 4,010 km (2,492 mi); the Hanford Site, 4,680 km (2,908 mi); and the Nevada Test Site, 4,172 km (2,592 mi). Distances along rail routes are slightly longer.

Ethnic and Income Characteristics: Figure 3-13 shows the ethnic composition for the area surrounding the port at Newport News. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 32 percent of the total population, and approximately 86 percent of the minority population for the area surrounding the port. Figure 3-14 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

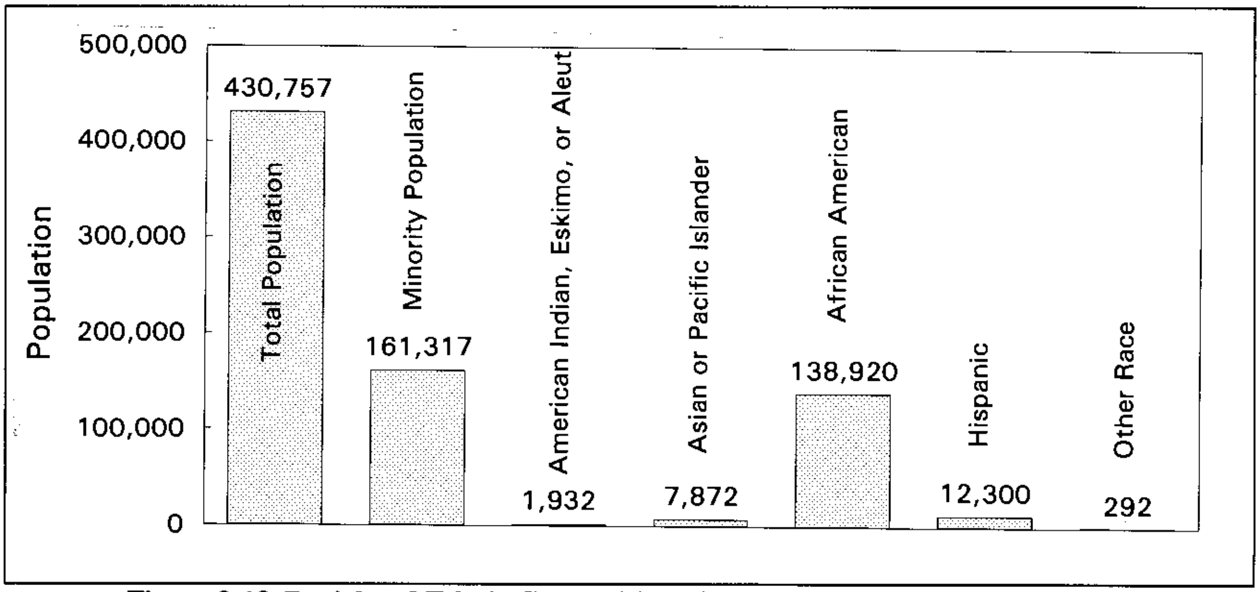


Figure 3-13 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Newport News

Norfolk International Terminal: This terminal is located on the south side of the Port in Norfolk, adjacent to the Navy Base on the Elizabeth River Channel. Norfolk International Terminal has 4 container vessel berths, 7 container cranes, a roll-on/roll-off berth, and covered pier storage. Sewell's Point Terminal, located at the north end (seaward) of Norfolk International Terminal's container berths has two piers, and covered storage for breakbulk cargoes. A map of Norfolk International Terminal is shown in Figure 3-15.

The 1990 population within 16 km (10 mi) of the port terminals was 681,864. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 131,000; the Oak Ridge Reservation, 174,000; the Idaho National Engineering Laboratory, 631,000; the Hanford Site, 694,000; and the Nevada Test Site, 694,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 800 km (497 mi); the Oak Ridge Reservation, 880 km (547 mi); the Idaho National Engineering Laboratory, 4,070 km (2,529 mi); the Hanford Site, 4,740 km (2,945 mi); and the Nevada Test Site, 4,240 km (2,635 mi). Distances along rail routes are slightly longer.

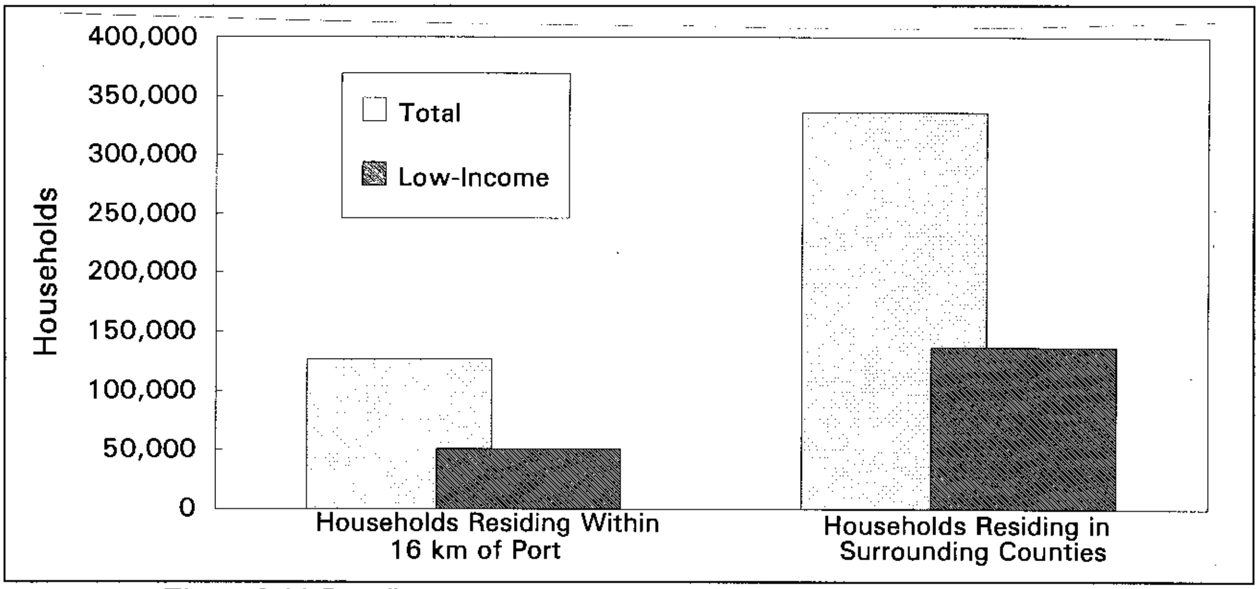


Figure 3-14 Low-Income Households Residing within 16 km (10 mi) of the Port of Newport News

Ethnic and Income Characteristics: Figure 3-16 shows the ethnic composition for the area surrounding the port at Norfolk, VA. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 33 percent of the total population, and approximately 93 percent of the minority population for the area surrounding the port. Figure 3-17 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

Portsmouth Marine Terminals: Portsmouth Marine Terminals are located at the confluence of the Elizabeth River and its western branch in the city of Portsmouth. The terminals have 3 berths that handle container, breakbulk and roll-on/roll-off cargoes. The terminals have 3 container cranes, and more than 14,000 m^2 (150,000 ft^2) of warehouse space. A map of the Portsmouth Marine Terminals is shown in Figure 3-18.

The 1990 population within 16 km (10 mi) of the Portsmouth Marine Terminals was 665,700. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 135,000; the Oak Ridge Reservation, 257,000; the Idaho National Engineering Laboratory, 670,000; the Hanford Site, 718,000; and the Nevada Test Site, 732,000. Populations along rail routes to these sites are about the same for eastern sites and slightly larger for western sites. The distances to the five potential sites on interstate routes are: the Savannah River Site, 810 km (503 mi); the Oak Ridge Reservation, 780 km (485 mi); the Idaho National Engineering Laboratory, 4,040 km (2,510 mi); the Hanford Site, 4,710 km (2,927 mi); and the Nevada Test Site, 4,210 km (2,616 mi). Distances along rail routes are slightly longer.

Ethnic and Income Characteristics: Figure 3-19 shows the ethnic composition for the area surrounding the port at Portsmouth. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 33 percent of the total population, and approximately 89 percent of the minority population for the area surrounding the port. Figure 3-20 shows analogous information for

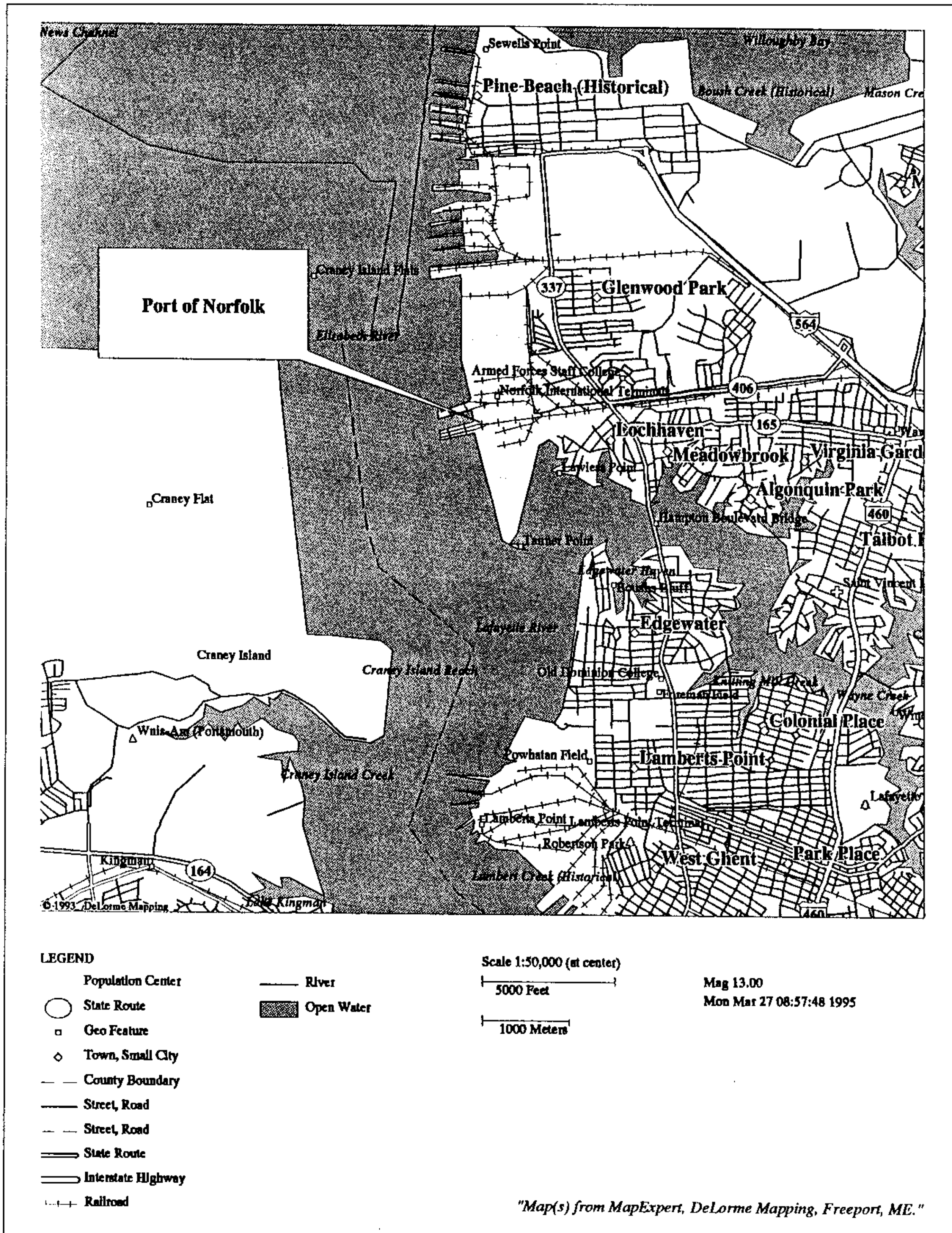


Figure 3-15 Port of Norfolk, VA

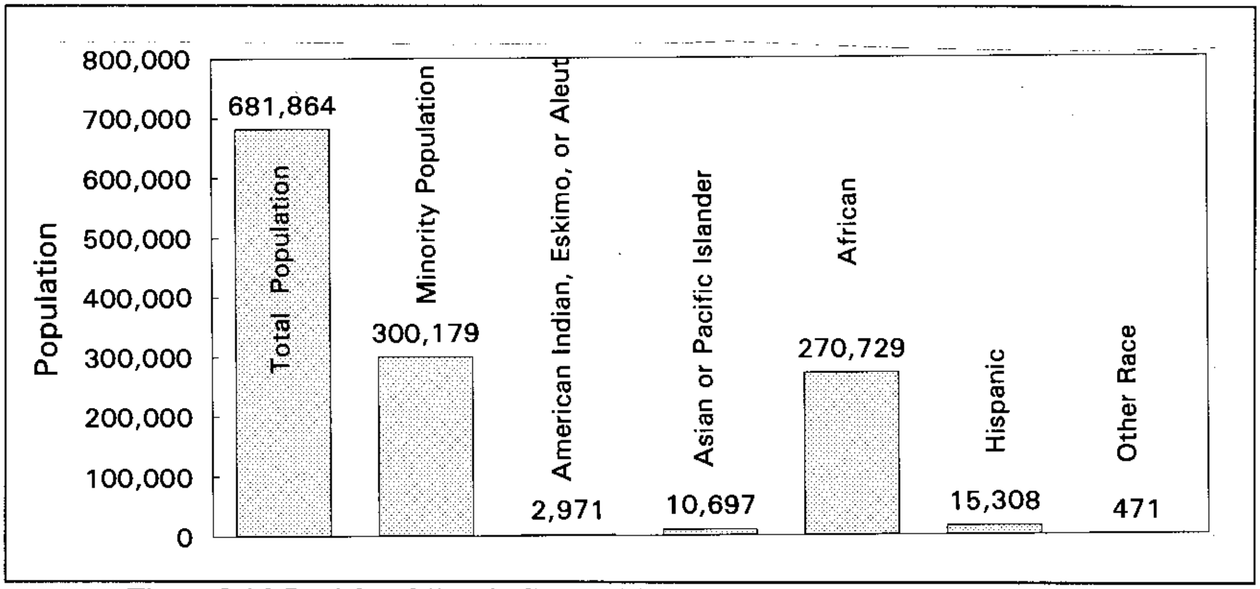


Figure 3-16 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Norfolk

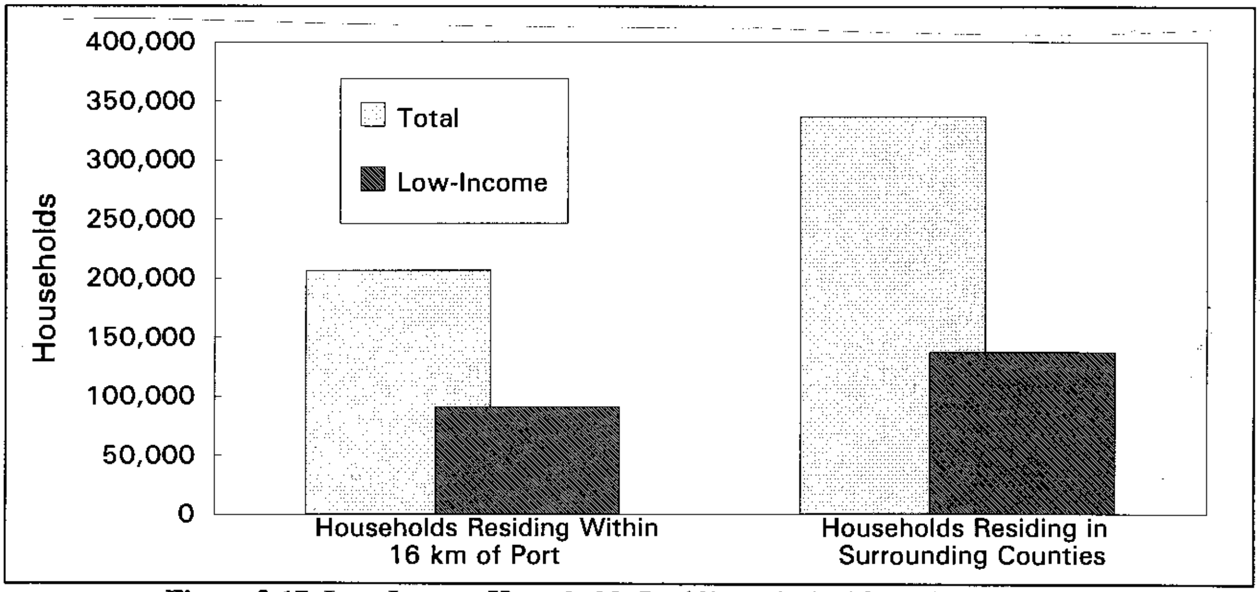


Figure 3-17 Low-Income Households Residing within 16 km (10 mi) of the Port of Norfolk

low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.4 Jacksonville, FL

The Port of Jacksonville is located on the Atlantic Coast of northern Florida, along the St. Johns River. It is a geographically large city (1,967 km^2 or 760 mi^2) ranging from the town of Orange on the east side of the river to Julington Creek on the west side. Most of the marine terminals are on the west side of the river, about 34 km (21 mi) from the ocean entrance. However, the Blount Island container terminal is well

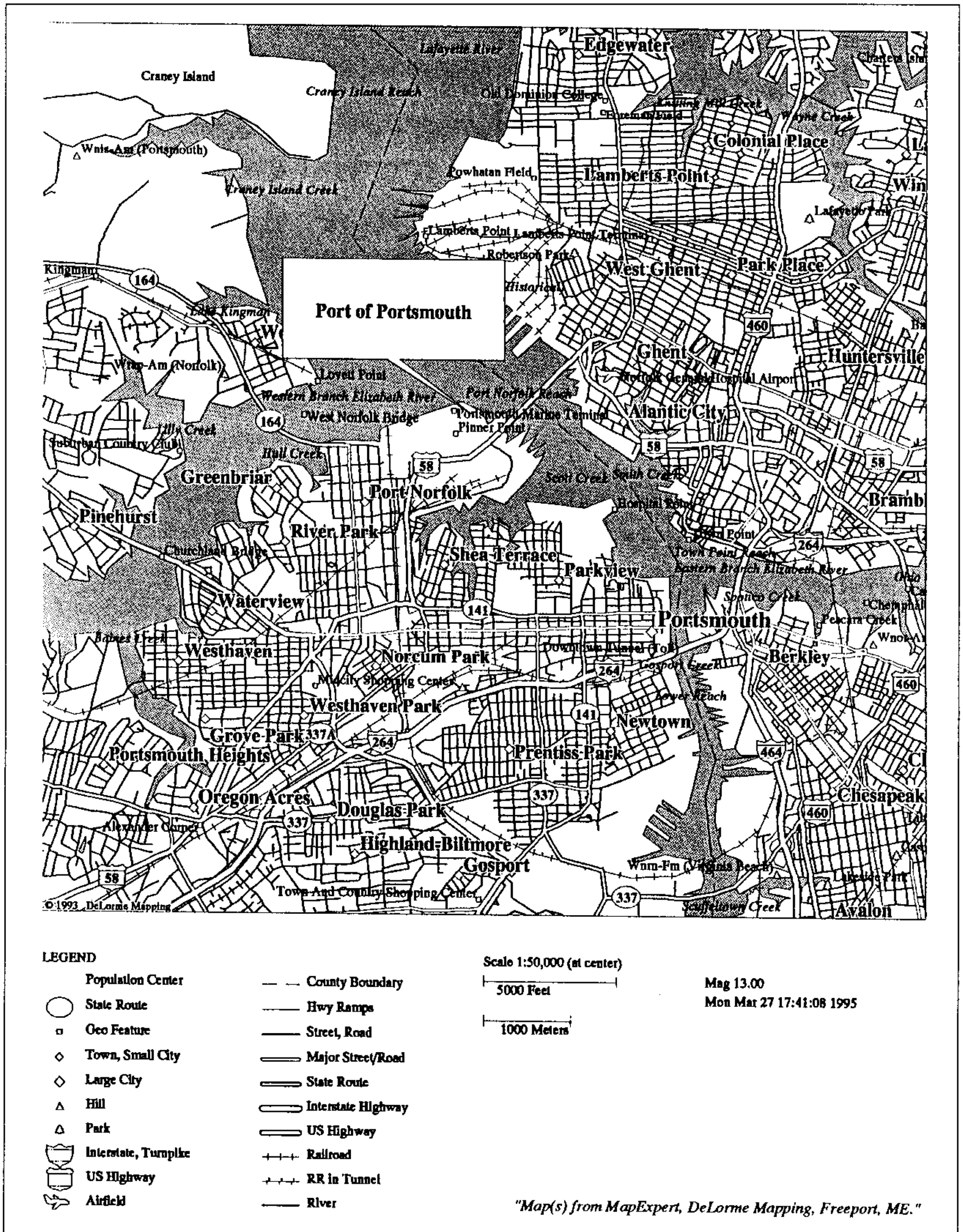


Figure 3-18 Port of Portsmouth, VA

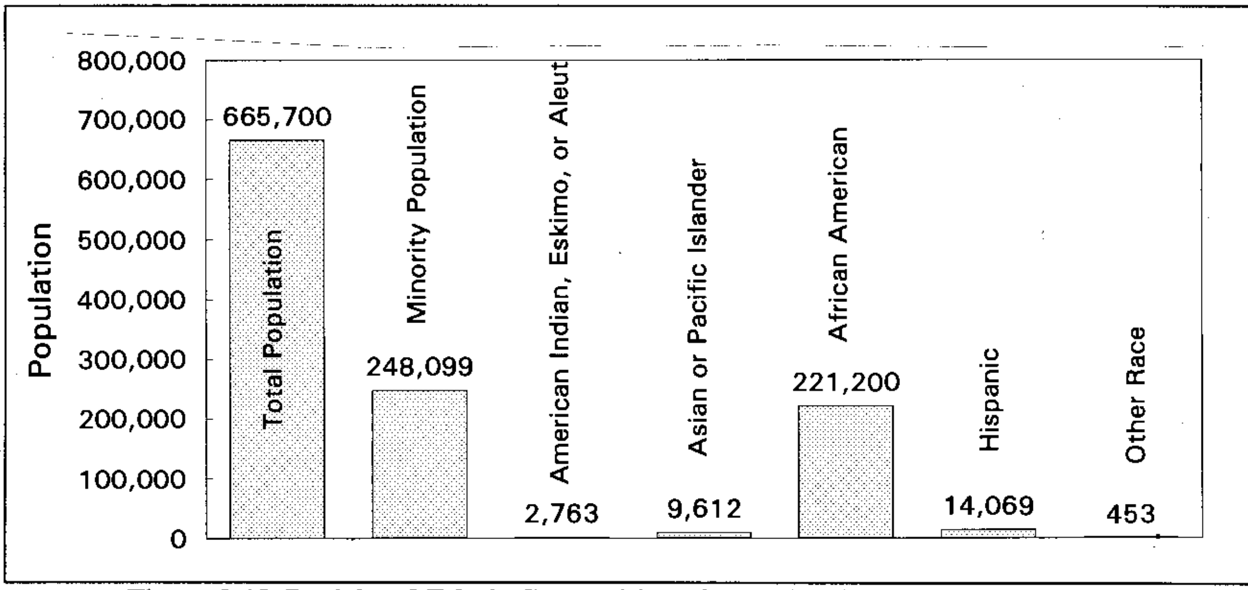


Figure 3-19 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Portsmouth

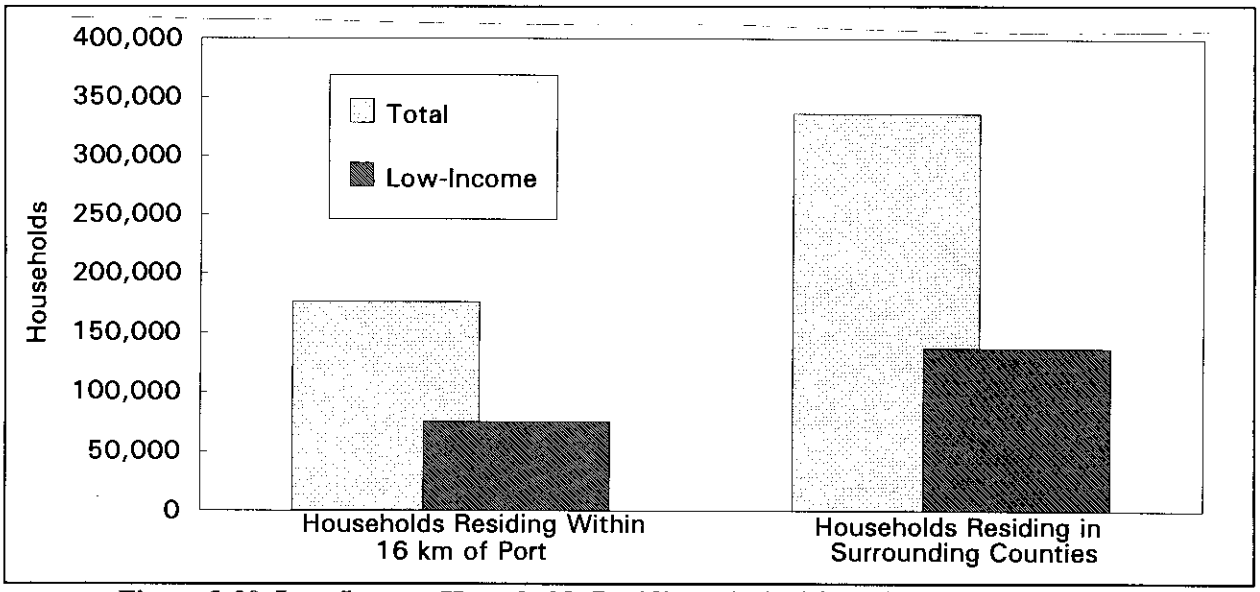


Figure 3-20 Low-Income Households Residing within 16 km (10 mi) of the Port of Portsmouth

separated from the city, and is only about 11 km (7 mi) from the harbor entrance. A map of the port is shown in Figure 3-21. A Federal project maintains a channel depth of 12.2 m (40 ft) to 12.8 m (42 ft) at the entrance to the river.

The St. Johns River has a deep, steep-sided channel cut through rock in some areas. Tidal currents are strong in the river as far as Jacksonville, approaching 3 knots in several places.

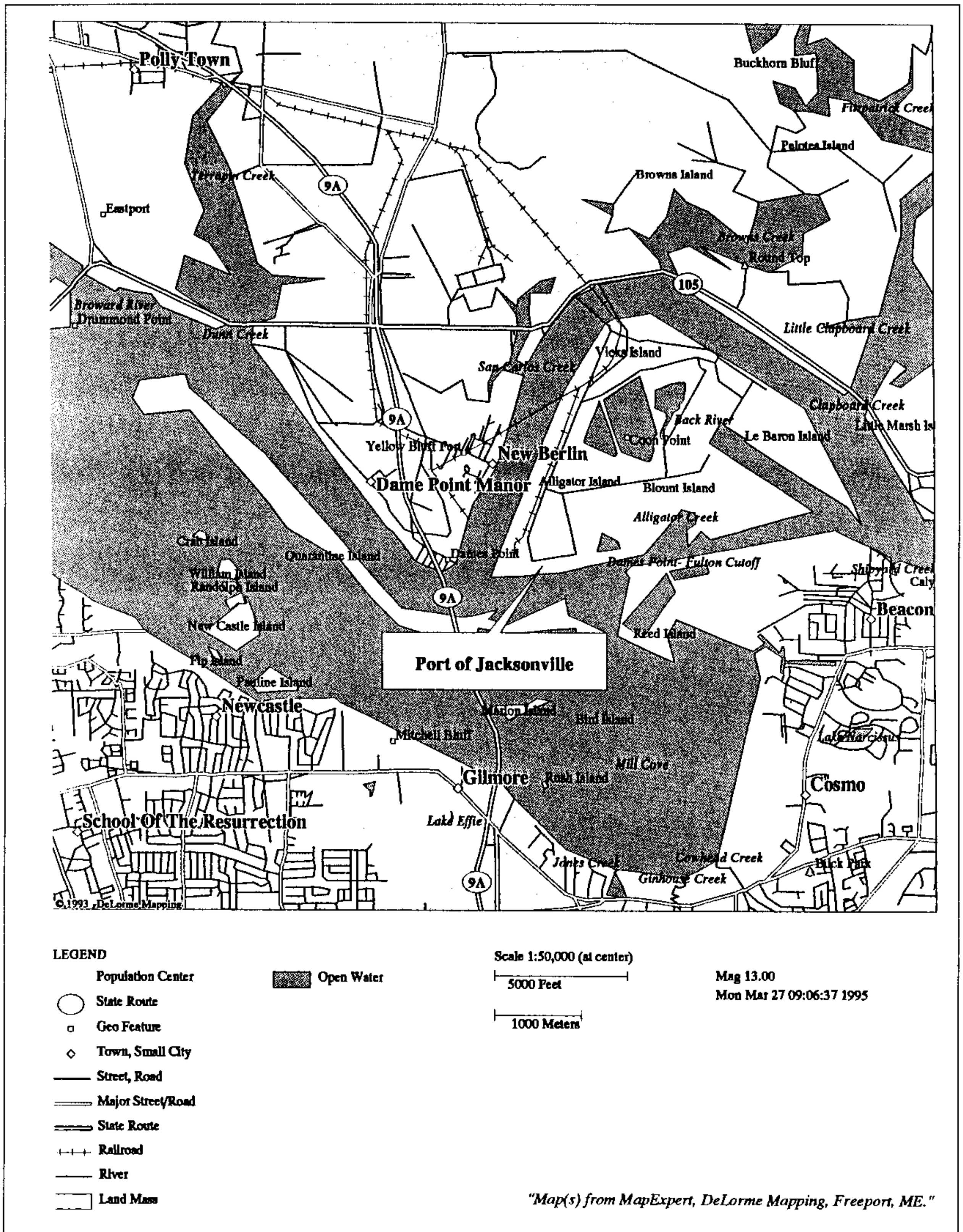


Figure 3-21 Port of Jacksonville, FL

There are two deepwater container/general cargo terminals: Blount Island, located approximately 11 km (7 mi) from the harbor entrance, and Talleyrand Docks and Terminals located about 34 km (21 mi) from the entrance. Both terminals are equipped with modern cranes, handle breakbulk and other types of cargo, and have warehouse as well as open storage areas. Of the two, Blount Island would be preferred because of its separation from the high-density downtown area and closer proximity to the sea.

Blount Island Terminal: Blount Island is a 356 ha (880 acre) facility with 1,920 m (6,336 ft) of berthing space. Blount Island berths 7-13 have 11.6 m (38 ft) of water alongside at mean low water, and five container cranes. This terminal is connected to the mainland via a fixed highway bridge which joins State Highway 105 (Necksher Drive) and connects with I-95 and Route 17 about five miles north of the city of Jacksonville. Blount Island has pierside service by the CSX Railroad, which connects with the Norfolk Southern Railroad.

Talleyrand Terminal: Talleyrand Docks is a 70 ha (173 acre) facility with 1,250 m (4,100 ft) of wharf on deep water (11.6 m or 38 ft at mean low water). It has two container cranes and two large gantry cranes. Talleyrand Terminal is located in downtown Jacksonville's shopping and commercial zone, about 2.9 km (1.8 mi) downstream of the John R. Matthews Bridge (alternate U.S. Route 90), and less than 1.0 km (0.6 mi) via city streets from the city Expressway system.

The 1990 population within 16 km (10 mi) of the port terminals was 334,212. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 46,900; the Oak Ridge Reservation, 175,000; the Idaho National Engineering Laboratory, 576,000; the Hanford Site, 643,000; and the Nevada Test Site, 639,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 607 km (377 mi); the Oak Ridge Reservation, 912 km (567 mi); the Idaho National Engineering Laboratory, 4,030 km (2,504 mi); the Hanford Site, 4,700 km (2,920 mi); and the Nevada Test Site, 4,190 km (2,604 mi). Distances along rail routes are about the same.

Environmental Conditions: The area between the mouth of the St. Johns River and Blount Island is characteristic of typical coastal lowlands found along the southeastern United States. Numerous creeks meander through large expanses of marshes and swamps. With the exception of the U.S. Naval Station Mayport and the village of Mayport, which occupy the first several miles along the southern bank of the river, the land bordering the lower portion of the river is largely undeveloped with the exception of riverfront residences, mainly along the northern bank. Most of the land to the north of the river between Blount Island and the coast is part of the Nassau River - St. Johns River Marshes Aquatic Preserve. The Fort Caroline National Memorial is located southeast of Blount Island on the southern bank of the river. The Little Talbot Island State Park is located approximately 1.6 km (1 mi) north of the channel entrance.

The lower 24.2 km (15 mi) of the St. Johns River has been designated as critical habitat for the manatee, a listed endangered species. The river is also used as a migratory area for the shortnose sturgeon, a listed endangered species (FWS, 1980b). According to the Florida Natural Areas Inventory, the following rare species have been reported within 3.2 km (2 mi) of the Blount Island Terminal: West Indian manatee (State and Federally Listed Endangered Species), shortnose sturgeon (State and Federally Listed Endangered Species), Atlantic sturgeon (State-Listed Species of Special Concern and Federally Listed Threatened Species), sea lamprey, and the opossum pipefish (Murray, 1994).

A variety of wading birds is also found in the vicinity of the Fort Caroline National Memorial. Several species of birds, including shorebirds, waterfowl, and gannets frequent the area around the jetties at the channel entrance. In particular, the brown pelican, a State Species of Special Concern, is found in this area. A variety of birds inhabits the Little Talbot Island State Park, including the American oystercatcher, a State Species of Special Concern. Loggerhead sea turtles, a listed endangered species, use the beaches along this portion of Florida as a nesting area (FWS, 1980b).

While environmental awareness is high throughout the state of Florida, there are no known sensitive wildlife sanctuaries in the immediate area of the Port of Jacksonville. Blount Island is surrounded by extensive marsh and wetlands.

Climatic Conditions: The Port of Jacksonville is located along the upper 39.4 km (24.5 mi) of the St. Johns River. The terrain in this area is relatively level, providing very little change in relief proceeding inland from the coastal region. The National Weather Service has been archiving meteorological information for this area since 1880.

The climate of this area is modified by the influence of the Atlantic Ocean. Easterly winds occur roughly 40 percent of the time, producing a true maritime climate for the Jacksonville area. The greatest rainfall occurs during summer, usually associated with afternoon and evening thunderstorms. During summer, measurable precipitation can be recorded nearly every two days. The prevailing winds are northeasterly in the fall and winter months, becoming more southwesterly during spring and summer. Although Jacksonville is along the eastern U.S. coast, it has been very fortunate in escaping hurricane-force winds. The majority of systems in recent years which have reached this latitude have moved parallel to the coastline, keeping well offshore. Others have weakened significantly moving overland prior to reaching metropolitan Jacksonville. The combination of these two factors has spared the area from any major devastation due to tropical systems in recent years (NOAA, 1992b). The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Jacksonville, the Uniform Building Code provides a basic wind speed of about 160 km per hour (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

Ethnic and Income Characteristics: Figure 3-22 shows the ethnic composition for the area surrounding the port at Jacksonville. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans were the largest minority group. Figure 3-23 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.5 Military Ocean Terminal Sunny Point, NC

The Military Ocean Terminal at Sunny Point (MOTSU) is a U.S. Department of Defense transportation facility located north of Southport, NC. The facility is located on the Cape Fear River, approximately 16 km (10 mi) upstream (north) from the mouth of the river, and 26 km (16.1 mi) south of the Port of Wilmington, NC. A map of MOTSU is shown in Figure 3-24. The port is easily accessed from the ocean, and all commercial vessels bound for Wilmington, NC must pass by MOTSU. It is served by a 12.1 m- (40 ft-) deep by 152 m- (500 ft-) wide channel from the ocean.

The water depth (channel and alongside the wharves) of 10.3 m (34 ft) at mean low water is adequate for most commercial breakbulk, roll-on/roll-off, and container ships. The terminal has three 600 m (2,000 ft) wharves, each with three berths. All wharves have three parallel sets of rail tracks. Berth 1, on the south

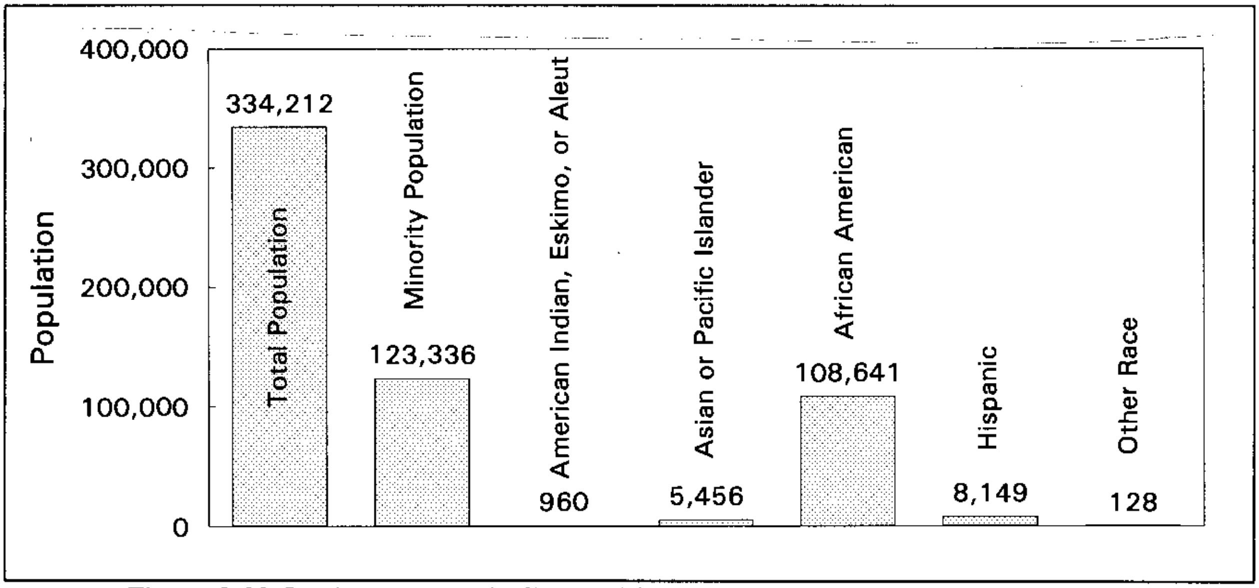


Figure 3-22 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Jacksonville

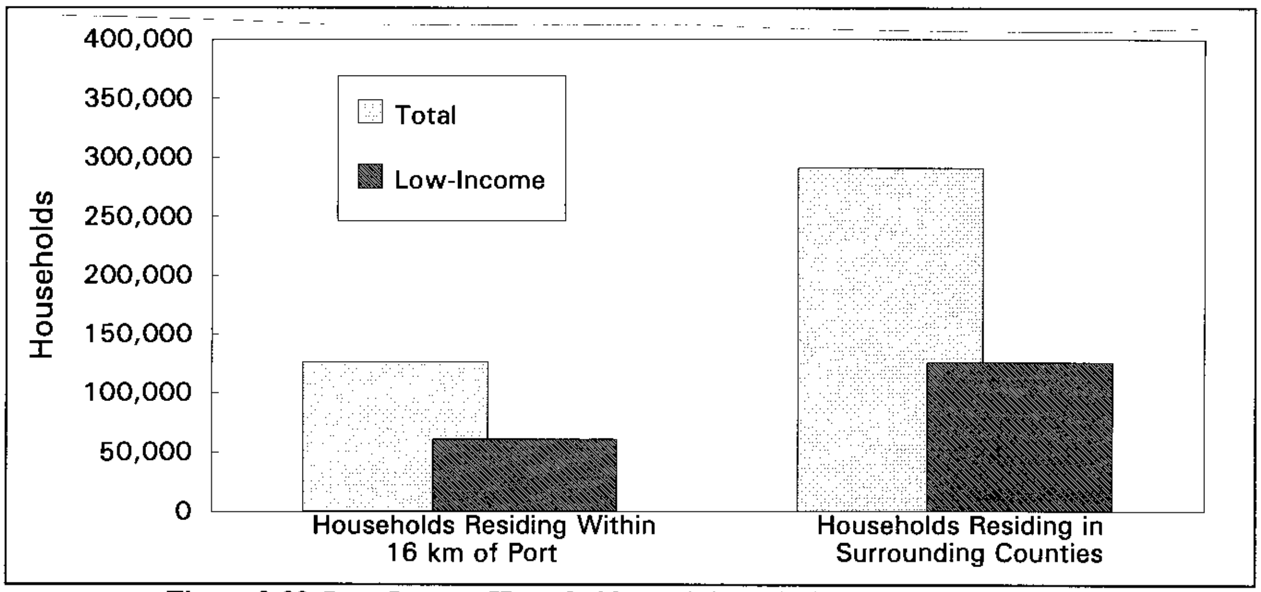


Figure 3-23 Low-Income Households Residing within 16 km (10 mi) of the Port of Jacksonville

wharf, has two 50 metric ton (55 ton) container cranes capable of off-loading container or container/breakbulk vessels. Berth 3 has been modified with a 30 m (100 ft) wide, reinforced concrete apron that permits breakbulk and roll-on/roll-off operations, in addition to containerized cargoes.

MOTSU is serviced by well-maintained roads, and has a dedicated 157 km (98 mi) U.S. Army rail line that connects the CSX network directly to the terminal. Truck access is provided by State Route 87 from the northwest and State Route 133 from the north. Route 87 provides access to U.S. 17, which runs southwest or northeast. The distance from the terminal gate to Route 133 is about 3.7 km (2.2 mi). Route 133 runs directly to U.S. 17 just outside Wilmington. From Wilmington, U.S. 74 runs west 120 km (75 mi) to Interstate 95, the nearest major north-south highway.

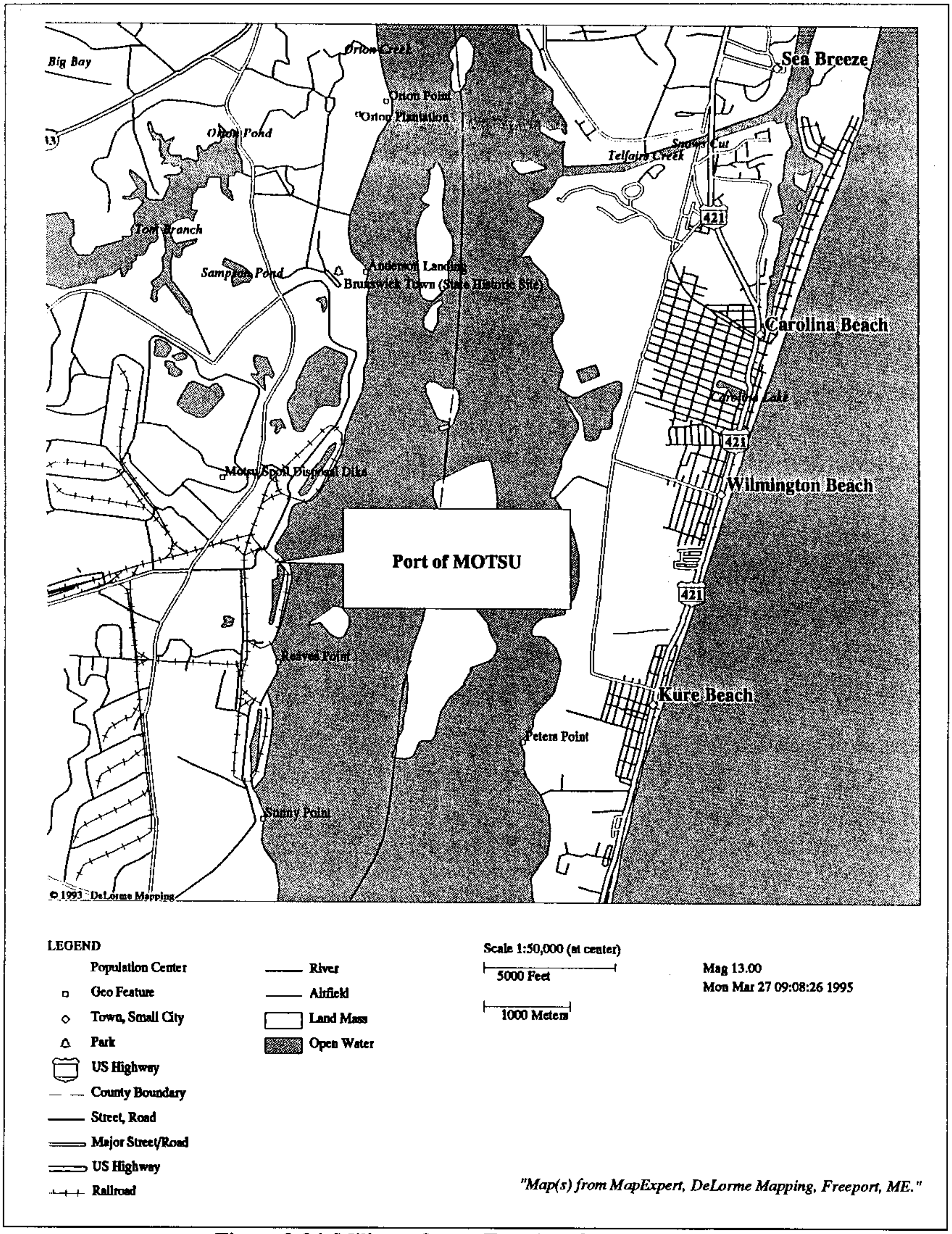


Figure 3-24 Military Ocean Terminal Sunny Point, NC

The 1990 population within 16 km (10 mi) of the port terminals was 7,995. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 34,200; the Oak Ridge Reservation, 128,000; the Idaho National Engineering Laboratory, 463,000; the Hanford Site, 548,000; and the Nevada Test Site, 619,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 402 km (250 mi); the Oak Ridge Reservation, 798 km (496 mi); the Idaho National Engineering Laboratory, 3,873 km (2,407 mi); the Hanford Site, 4,615 km (2,868 mi); and the Nevada Test Site, 3,953 km (2,456 mi). Distances along rail routes are slightly longer.

Environmental Conditions: The environmental conditions at MOTSU are similar to those at the Port of Wilmington, NC, and are described in Section 3.2.1.10. MOTSU has been identified as an area with sinkhole activities (Koch, 1984). Sinkholes can occur naturally or as a result of human activity. Occurrences of sinkholes are closely tied to the drainage pattern in areas where geologic formations provide a collapse mechanism. Human activities which modify the natural drainage pattern in such an area can increase the rate of sinkhole formation. Sinkholes pose a potential hazard to truck and rail traffic in the Sunny Point area. Due to the robust casks which would be used to transport spent nuclear fuel from foreign research reactors (see Appendix B for a description of the casks), sinkholes would not be expected to cause a radiological accident.

Climatic Conditions: The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. For MOTSU, the Uniform Building Code provides a basic wind speed of about 160 km per hour (100 mph) (UBC, 1991). The port is located in a low seismic zone with an acceleration of 0.075 g.

Other climatic conditions at MOTSU are similar to those at the Port of Wilmington, NC, and are described in Section 3.2.1.10.

Ethnic and Income Characteristics: Figure 3-25 shows the ethnic composition for the area surrounding MOTSU. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 17 percent of the total population, and approximately 91 percent of the minority population for the area surrounding the port. Figure 3-26 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.6 Naval Weapons Station, Concord, CA

Naval Weapons Station (NWS) Concord is located on the southern edge of Suison Bay, an estuarine area immediately west of the junction of the Sacramento and San Joaquin Rivers. By water, the NWS is approximately 58 km (36 mi) northeast of the Golden Gate Bridge. The city of Concord, CA, is about 8 km (5 mi) south of the NWS. A map of the NWS military port is shown in Figure 3-27.

The 1990 population within 16 km (10 mi) of the port terminal was 381,070. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 1,040,000; the Oak Ridge Reservation, 742,000; the Idaho National Engineering Laboratory, 271,000; the Hanford Site, 263,000; and the Nevada Test Site, 437,000. Populations along rail routes to these sites are slightly smaller for the Oak Ridge Reservation, the Idaho National Engineering Laboratory and the Nevada Test Site, and slightly larger for the Savannah River Site and the Hanford Site. The distances to the five potential sites on interstate routes are: the Savannah River Site, 4,476 km (2,781 mi); the Oak Ridge Reservation, 4,111 km (2,554 mi); the Idaho National Engineering Laboratory,

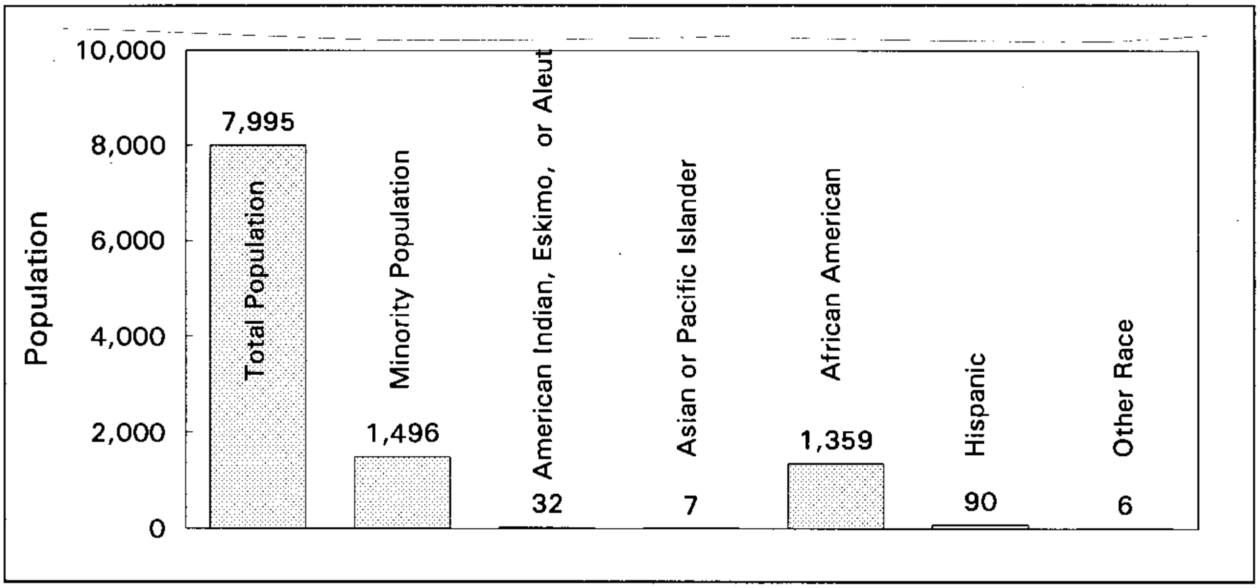


Figure 3-25 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of MOTSU

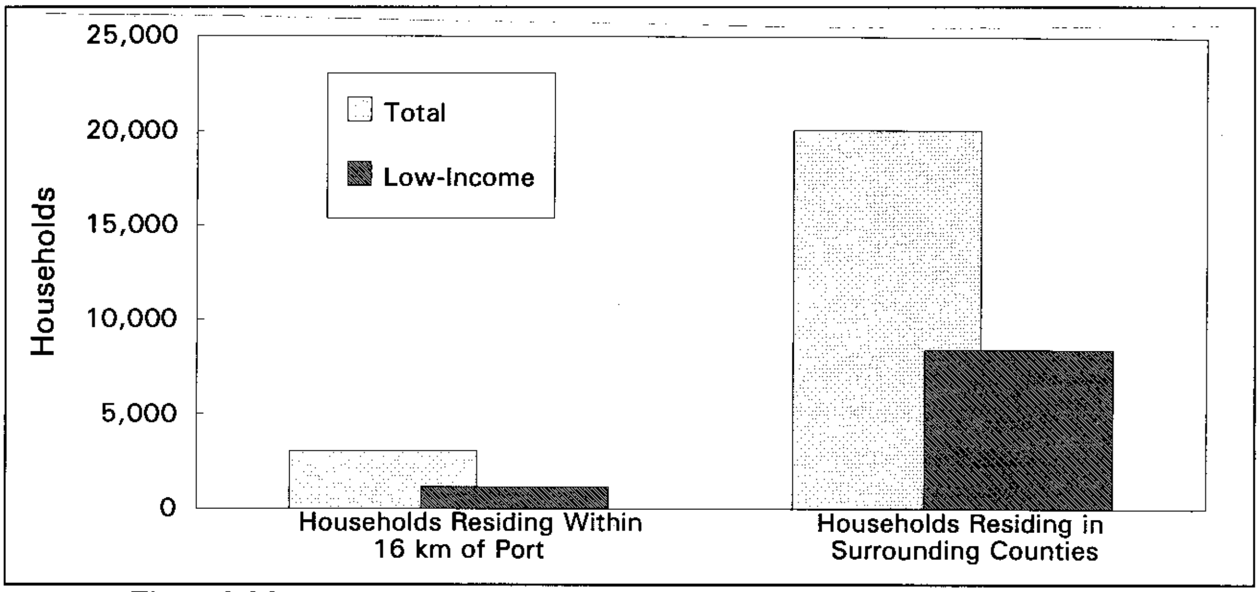


Figure 3-26 Low-Income Households Residing within 16 km (10 mi) of MOTSU

1,516 km (942 mi); the Hanford Site, 1,376 km (855 mi); and the Nevada Test Site, 1,145 km (711 mi). Distances along rail routes are about the same for the Idaho National Engineering Laboratory, and slightly longer for the Savannah River Site, the Oak Ridge Reservation, the Hanford Site, and the Nevada Test Site.

Environmental Conditions: NWS Concord occupies 5,233 ha (12,931 acres) of land adjoining south Suison Bay. Of this total acreage, 2,135 ha (5,276 acres) are inland, while 3,097 ha (7,653 acres) are more tidal in nature. Wetlands comprise approximately 1,215 ha (3,002 acres) of the tidal area (Yocum, 1994). Wetlands occupy large areas of land bordering all sides of Suison Bay and Grizzly Bay, which is located

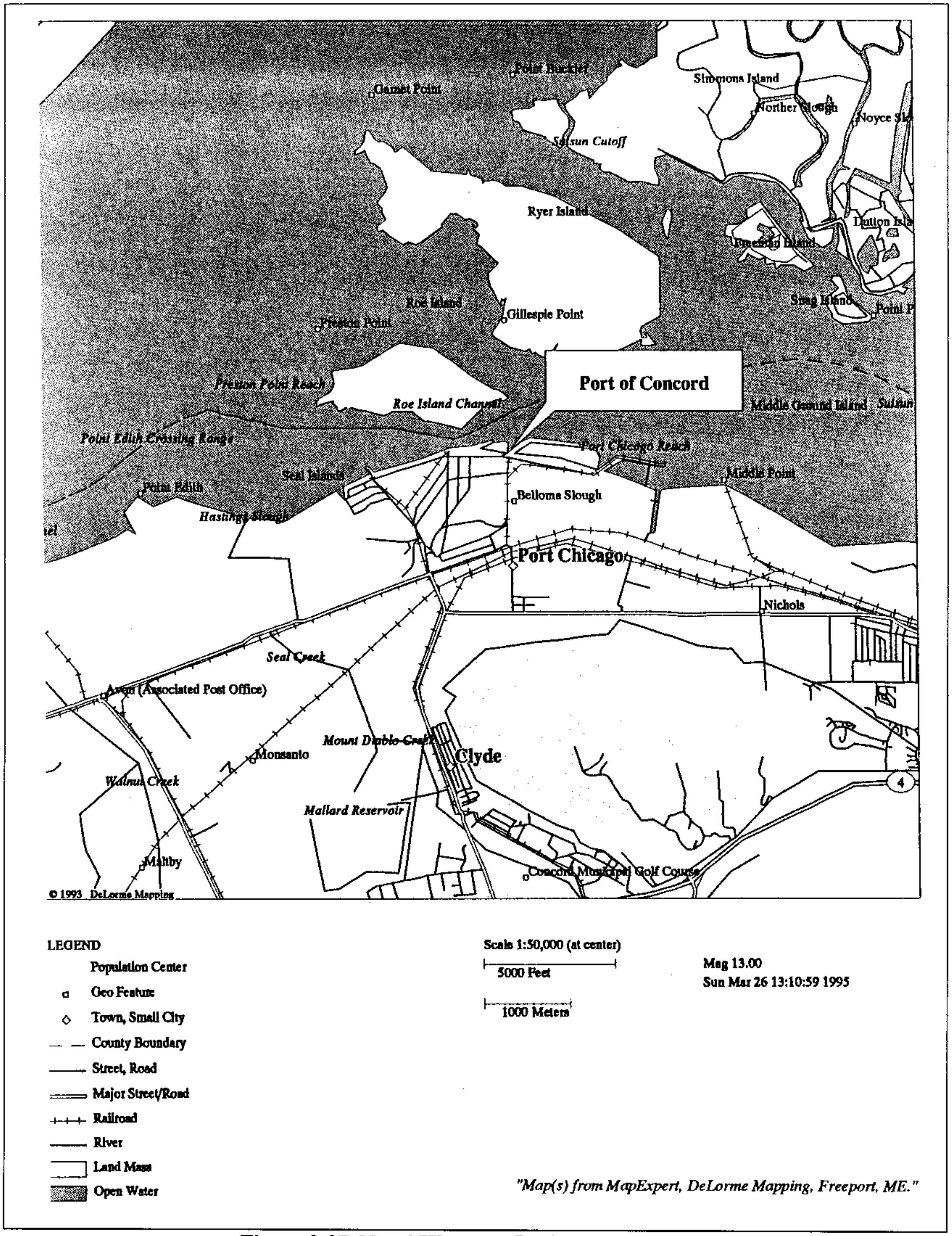


Figure 3-27 Naval Weapons Station Concord, CA

directly north of Suison Bay. The waters of Suison Bay are characterized as a mid-salinity estuarine habitat (generally 5 to 16.5 ppt). Chinook salmon, steelhead trout, striped bass, sturgeon, and American shad are typically found in this area (FWS, 1981a; FWS, 1981b).

Portions of the inland area at NWS Concord serve as a sanctuary for Tule elk, a formerly endangered species (Yocum, 1994). Other terrestrial species found in the area include the river otter, the salt-marsh harvest mouse (a Federally protected species), and the white-tailed kite. Adult concentrations and nesting areas of the California clapper rail (a Federally protected bird species) and the California black rail (a State protected species) are also found in this area. The Federally and State protected figwort plant family is also found in the vicinity of NWS Concord. In general, the greater San Francisco Bay area annually supports large numbers of shorebirds, wintering waterfowl, raptors, seabirds, and passerlings. In addition, shorebirds, wading birds, waterfowl, seabirds, and songbirds migrate through this coastal area.

Climatic Conditions: The climate is mild, with plenty of sunshine year round. Cloudless skies prevail during the spring, summer, and fall. Winter is the rainy season. Snow is rare, as are freezing temperatures. Sometimes torrential rains on the slopes can cause flooding along the Sacramento River (NOAA, 1993b).

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Concord NWS, the Uniform Building Code provides a basic wind speed of about 110 km per hour (70 mi per hour). The port is located on the edge of a very high seismic zone with an acceleration of 0.45 g.

Ethnic and Income Characteristics: Figure 3-28 shows the ethnic composition for the area surrounding NWS Concord. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 7 percent of the total population, and approximately 24 percent of the minority population for the area surrounding the port. Other minorities include Asian or Pacific Islanders (11 percent of total population), and Hispanics (10 percent of total population). These groups constitute 38 percent and 35 percent of the minority population, respectively. Figure 3-29 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.7 Portland, OR

The Port of Portland is located about 160 km (100 mi) above the mouth of the Columbia River on the Willamette River tributary. Portland is the principal city of the Columbia River system, and one of the major ports on the Pacific Coast. The container terminal that would be used for potential receipt of foreign research reactor spent nuclear fuel is located approximately 170 km (106 mi) from the entrance of the Columbia River. Federal project depths in the Columbia River are 14.6 m (48 ft) at the mouth of the river, and 12 m (40 ft) at Portland. A map of the port is shown in Figure 3-30.

The 1990 census population within 16 km (10 mi) of the port was 356,064. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 686,000; the Oak Ridge Reservation, 519,000; the Idaho National Engineering Laboratory, 143,000; the Hanford Site, 85,700; and the Nevada Test Site, 375,000. Populations along rail routes to these sites are slightly smaller for the Nevada Test Site and the Idaho National Engineering Laboratory, but slightly larger for the Savannah River Site, the Oak Ridge Reservation, and the Hanford

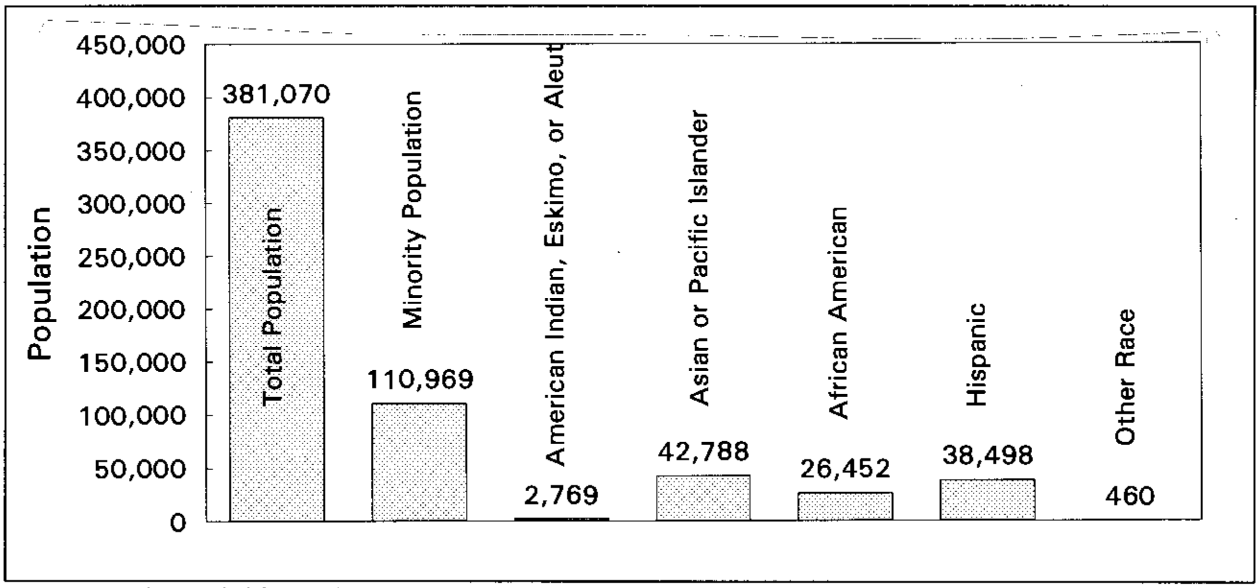


Figure 3-28 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of NWS Concord

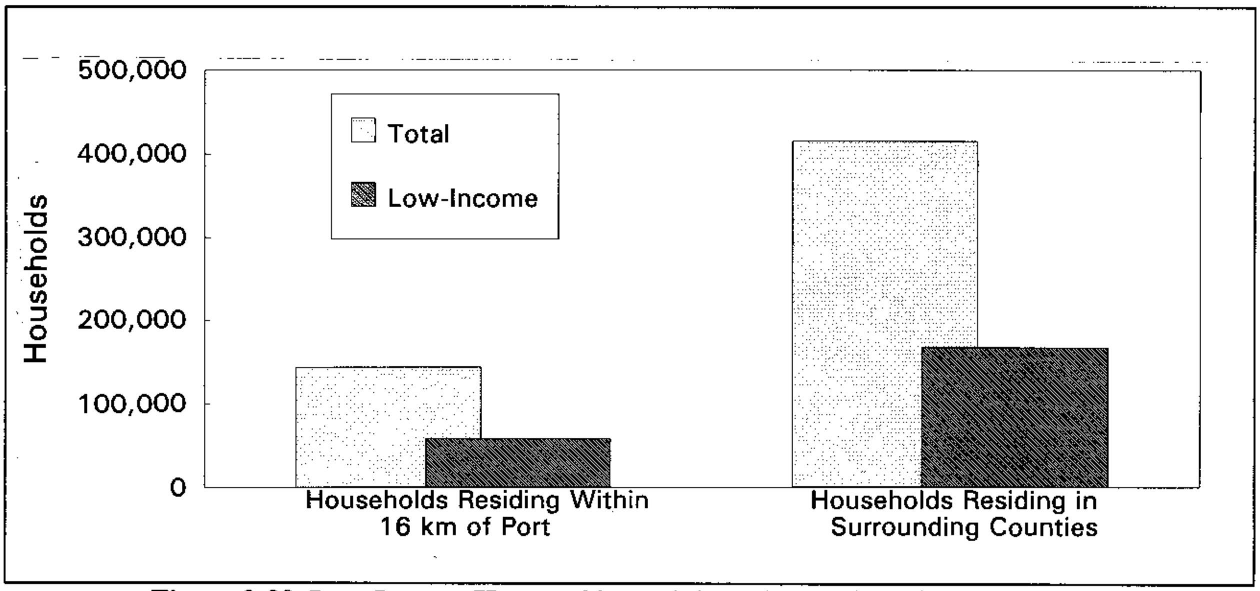


Figure 3-29 Low-Income Households Residing within 16 km (10 mi) of NWS Concord

Site. The distances to the five potential sites on interstate routes are: the Savannah River Site, 4,630 km (2,877 mi); the Oak Ridge Reservation, 4,200 km (2,610 mi); the Idaho National Engineering Laboratory, 1,190 km (739 mi); the Hanford Site, 407 km (253 mi); and the Nevada Test Site, 2,040 km (1,268 mi). Distances along rail routes are slightly longer, with the exception of the Hanford Site, which is slightly shorter.

Environmental Conditions: There are no known areas of special environmental concern in the immediate vicinity of the port, although concern for the environment runs high throughout the Pacific Northwest. The areas surrounding the Terminal are in river-oriented industrial land use. Wildlife habitat along the

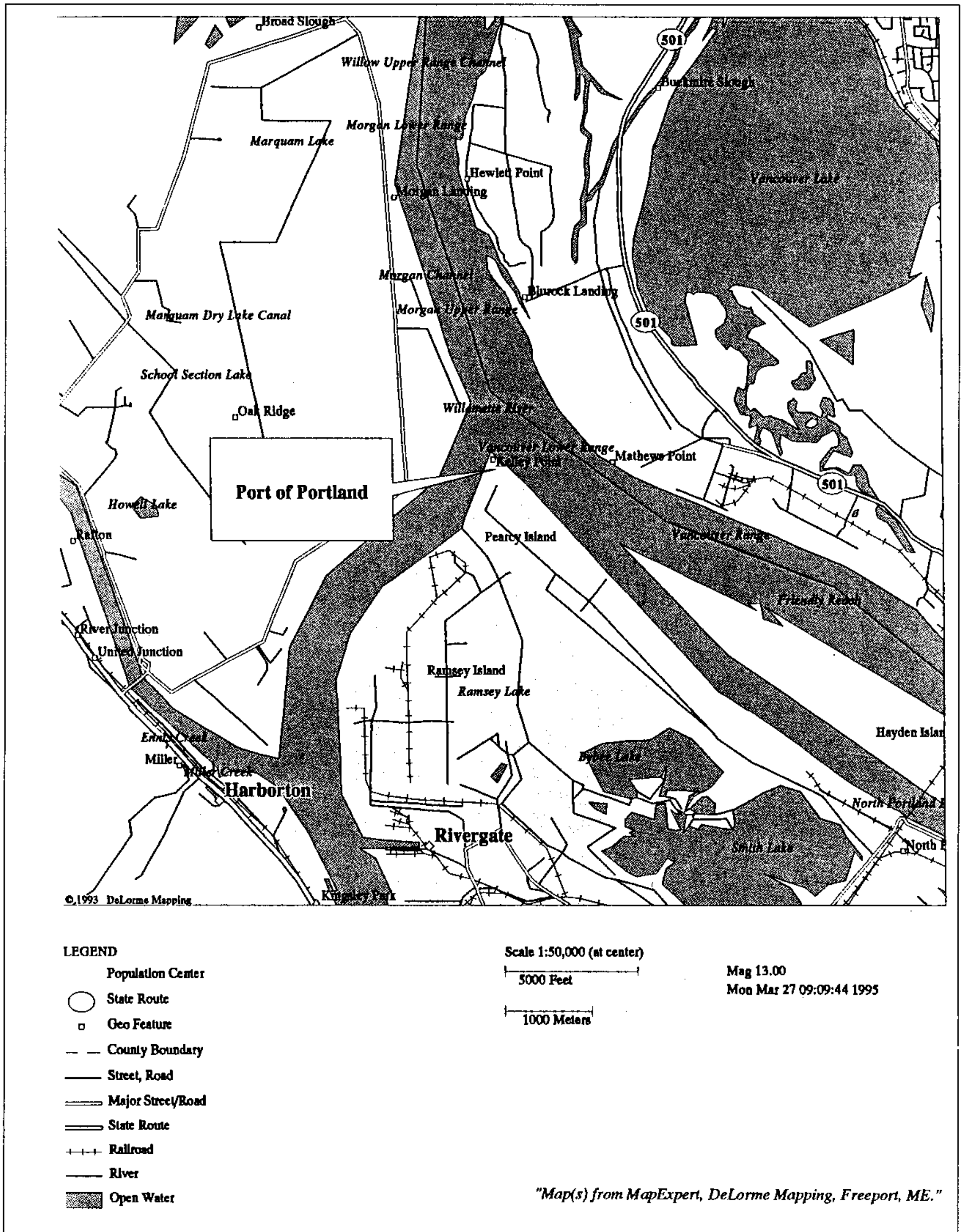


Figure 3-30 Port of Portland, OR

Oregon Slough is limited because of the industrial development, although some waterfowl use the area. While the primary uses in the Terminal area are commercial navigation and industry, some recreational fishing and boating occurs in Oregon Slough and the Columbia River.

The U.S. Fish and Wildlife Service's Ecological Inventory for the Vancouver, Washington-Oregon area indicates that the Columbia River generally includes the following fish species: salmonids, chinook salmon, coho salmon, chum salmon, pink salmon, sockeye salmon, steelhead trout, Dolly Varden, smelts, river lamprey, white sturgeon, American shad, eulachon and cutthroat trout (FWS, 1981c). South of Portland, the various islands and wetlands along the Columbia River provide habitat for a wide variety of terrestrial organisms. Areas of special interest include the Sauvie Island Game Management Area, which is located approximately 8 km (5 mi) downriver of Terminal 6, and the Ridgefield National Wildlife Refuge, which is approximately 16 km (10 mi) downriver.

The U.S. Army Corps of Engineers reports that raptors such as the red-tail hawk, bald eagle, and peregrine falcon are occasional visitors to this area and the U.S. Fish and Wildlife Service has indicated that the endangered American peregrine falcon and threatened bald eagle may winter in this area. In addition, the National Marine Fisheries Service has listed the Snake River sockeye salmon as endangered, and two Snake River chinooks stocks as threatened (Kurkoski, 1994). The State of Oregon's Natural Heritage Program reports that there are at least two rare species that occur in the vicinity of Terminal No. 6 (Gaines, 1994). These species are the painted turtle (a State-Sensitive-Critical species) and the Columbia water-meal.

Climatic Conditions: The port area is subject to earthquakes and volcanism (NOAA, 1992d). The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. For the Port of Portland, the Uniform Building Code provides a basic wind speed of about 140 km per hour (90 mph) (UBC, 1991). The port is located in a moderate seismic zone with an acceleration of 0.20 g. There have been two major earthquakes in the Puget Sound area this century (Bolt, 1978). On May 18, 1980, nearby Mount St. Helens suffered a major volcanic eruption. All the mountains along the Cascade Range are volcanic in origin and have some potential for eruption.

Ethnic and Income Characteristics: Figure 3-31 shows the ethnic composition for the area surrounding the port at Portland. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 8 percent of the total population, and approximately 50 percent of the minority population for the area surrounding the port. Hispanics and Asian or Pacific Islanders each accounted for about 20 percent of the minority population. Figure 3-32 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.8 Savannah, GA

The Port of Savannah is located on the south bank of the Savannah River, about 35 km (22 mi) above the entrance from the Atlantic Ocean. Savannah is the third largest city in Georgia, and is the chief port of the State of Georgia. The Savannah River serves as the boundary between Georgia and South Carolina. There are three large cargo terminals at the port. One of these terminals, Containerport, is a dedicated container handling terminal. It is located about 45.6 km (28.3 mi) up the Savannah River from the Atlantic Ocean. A map of the port is shown in Figure 3-33.

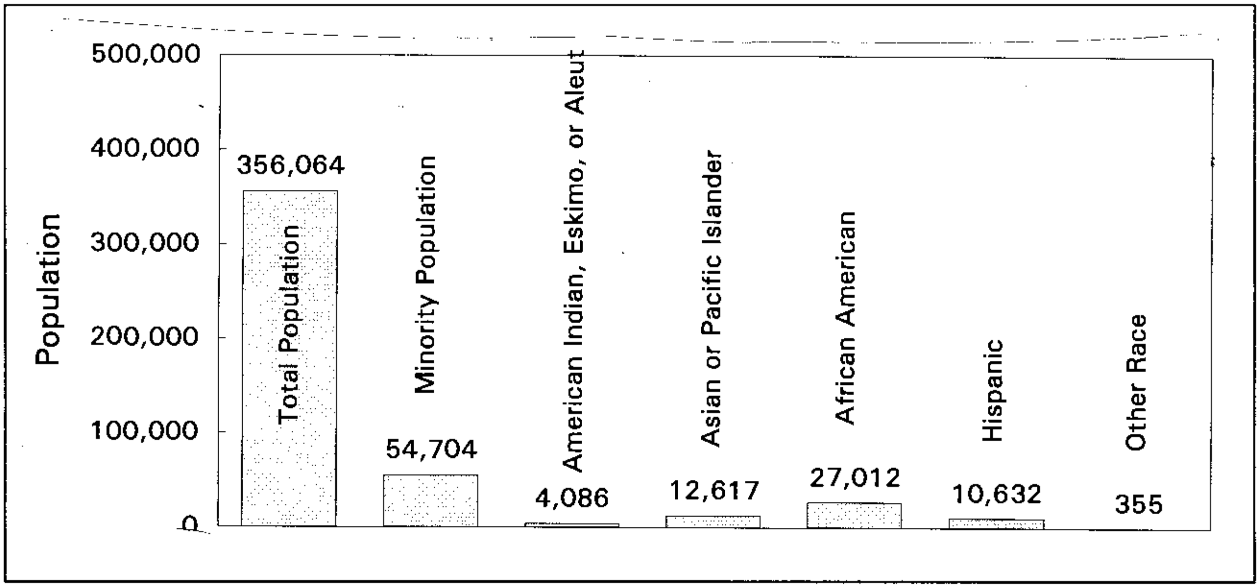


Figure 3-31 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Portland

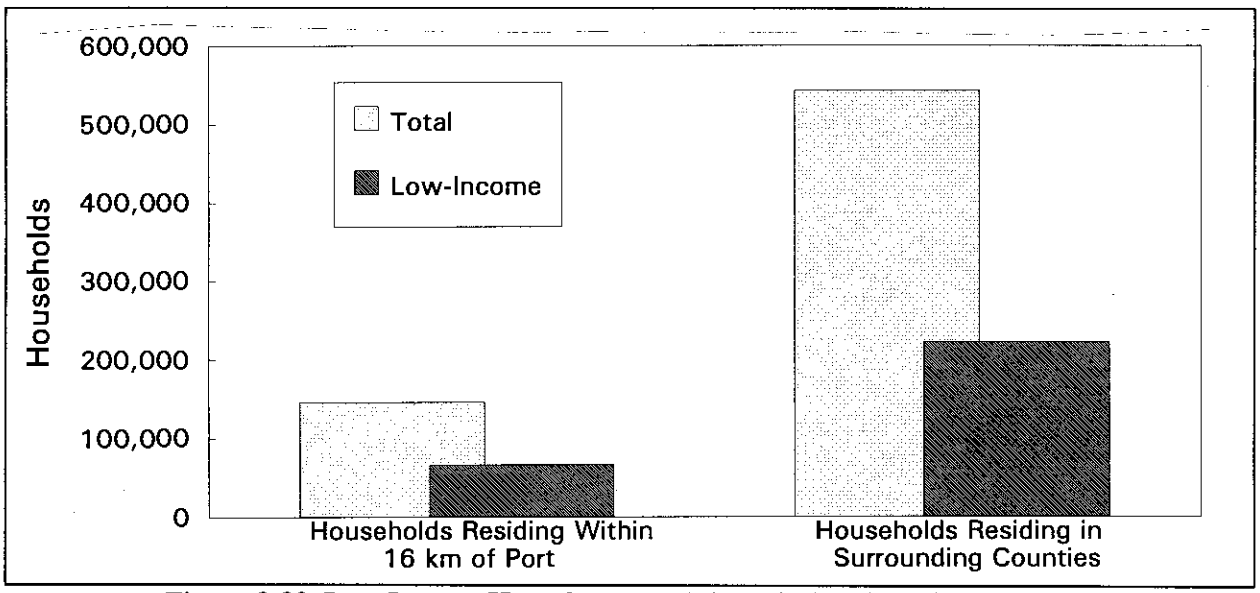


Figure 3-32 Low-Income Households Residing within 16 km (10 mi) of the Port of Portland

The 1990 population within 16 km (10 mi) of the port terminals was 155,166. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 37,300; the Oak Ridge Reservation, 101,000; the Idaho National Engineering Laboratory, 553,000; the Hanford Site, 602,000; and the Nevada Test Site, 616,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 400 km (249 mi); the Oak Ridge Reservation, 720 km (447 mi); the Idaho National Engineering Laboratory, 3,860 km (2,398 mi); the Hanford Site, 4,530 km (2,815 mi); and the Nevada Test Site, 4,020 km (2,498 mi). Distances along rail routes are slightly longer.

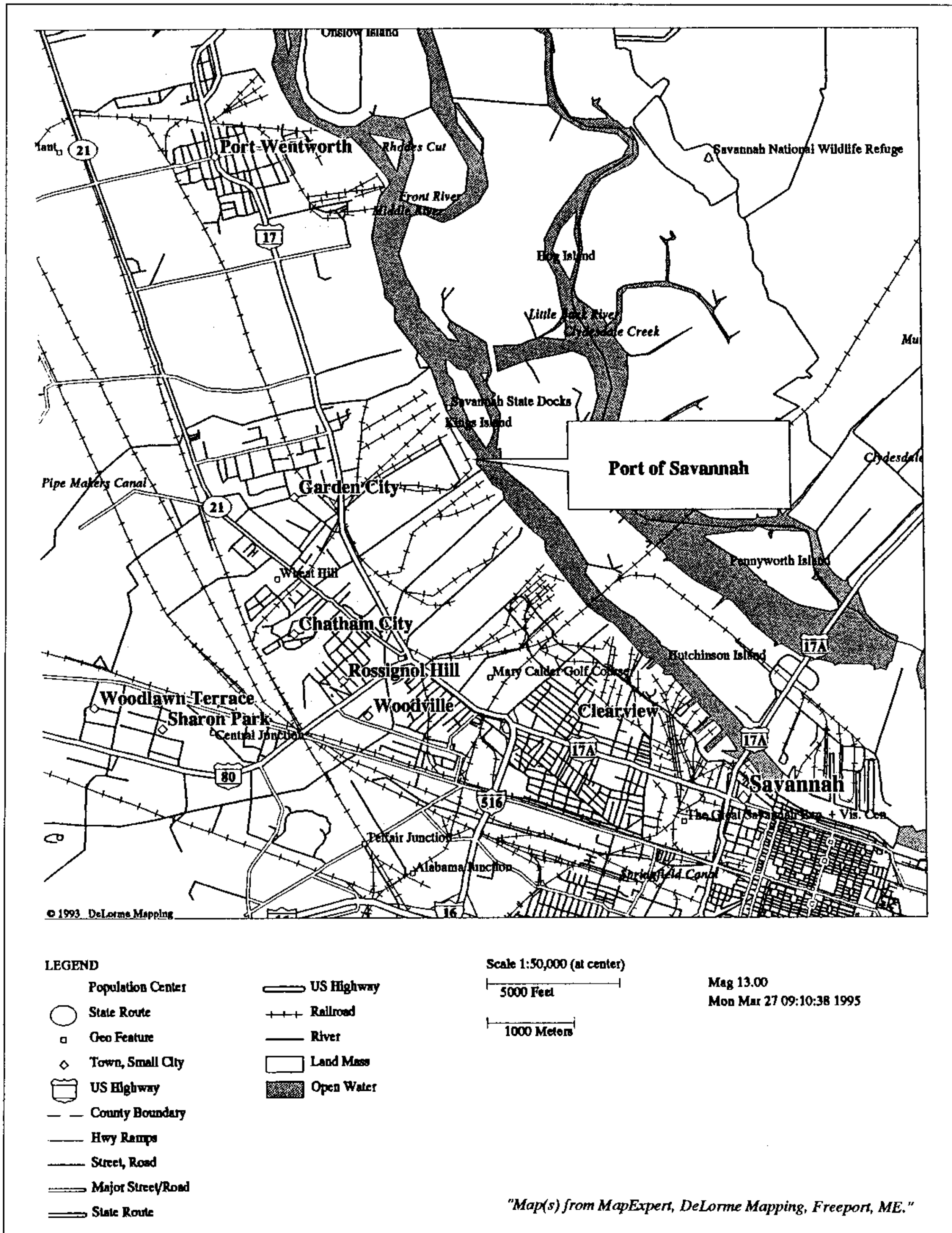


Figure 3-33 Port of Savannah, GA

Environmental Conditions: The lower Savannah River has multiple branches that meander through a variety of coastal lowlands including salt marshes, tidal creeks, freshwater marshes, and freshwater impoundments. South Carolina has classified its portion of the Savannah harbor upstream from Fort Pulaski (located at the mouth of the Savannah River) as Class B, and the portion oceanward as Class SA. Class B waters are freshwaters suitable for secondary contact recreation, use as a drinking water source following conventional treatment, fishing, industrial, and agricultural use. Class SA waters are defined as tidal salt waters suitable for primary contact recreation, and for all the uses listed in Class B. The State of Georgia has classified the Savannah River from mile 0 at Fort Pulaski north to mile 5 at Field's Cut as recreation waters. North of Field's Cut, the waters are classified as Coastal Fishing (U.S. Army, 1991). The river in the vicinity of Containerport is characterized as a transitional estuarine habitat, where the salinity ranges from low (generally 0.5 to 5 ppt) to mid-salinity (generally 5 to 16.5 ppt) (FWS 1980c).

A large number of aquatic and terrestrial species are found in and around the Savannah River near Containerport. State or Federally protected, endangered, or threatened aquatic species in the vicinity of Containerport include the Shortnose sturgeon and the Florida manatee, both identified as State and Federally endangered species. The Shortnose sturgeon uses the Savannah River as a migratory area. In addition, the Loggerhead turtle, the Bald eagle, and the American alligator are found along the lower reaches of the Savannah River (FWS, 1980c).

Both invertebrate and fish species of commercial and recreational value are found in the Savannah River. Commercial fishing is primarily for the American shad, sturgeon, shrimp, and blue crab. Public shellfishing is allowed in some areas near the mouth of the Savannah River, in the vicinity of Fort Pulaski. The Savannah River hosts the migration of several important commercial and game fishes, including the American shad, the hickory shad, and the blueback herring. Game species include the spotted seatrout, red drum, croaker, spot, striped bass, flounder, silver perch, white catfish, channel catfish, largemouth bass, sunfish, and crappies. The State of Georgia has closed the striped bass fishery for population recovery purposes (Schmitt, 1993).

There are several wildlife refuges and/or game management areas located along the lower portion of the Savannah River. Tybee National Wildlife Refuge is located at the mouth of the Savannah River at the confluence with the Atlantic Ocean. Just north of Tybee National Wildlife refuge is the Turtle Island Game Management Area. Containerport itself is located across the river from the southern end of the 10,371 ha (25,627 acre) Savannah National Wildlife Refuge. The Savannah National Wildlife Refuge and the Tybee National Wildlife Refuge are managed by the U.S. Fish and Wildlife Service.

Climatic Conditions: The area has a temperate climate, which is greatly influenced by winds coming into the area off the ocean. Nominally, 50 percent of the rainfall occurs during thunderstorms, with the remainder being equally distributed over the year and generally related to weather front passages. Severe tropical systems affect the Savannah area roughly once every 10 years and cause heavy, sustained precipitation, high winds, and extreme, but usually localized, coastal flooding. Rainfall measurements in excess of 51 cm (20 in) have been observed as a result of tropical systems impacting the area. Based on the 1951-1980 climatology, the first freeze occurs on average around November 15, and the last near March 10 (NOAA, 1992e).

The Port is subject to severe hurricanes and tropical storms, and given its proximity to Charleston, SC may have a slightly higher risk of earthquakes than the rest of the State of Georgia. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Savannah, the Uniform Building code provides a basic wind speed of about 130 km per hour (80 mi per hour). The port is located in a low seismic zone with an acceleration of 0.075 g.

Ethnic and Income Characteristics: Figure 3-34 shows the ethnic composition for the area surrounding the port at Savannah. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 49 percent of the total population, and approximately 95 percent of the minority population for the area surrounding the port. Figure 3-35 shows analogous information for low-income households residing within 16 km of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

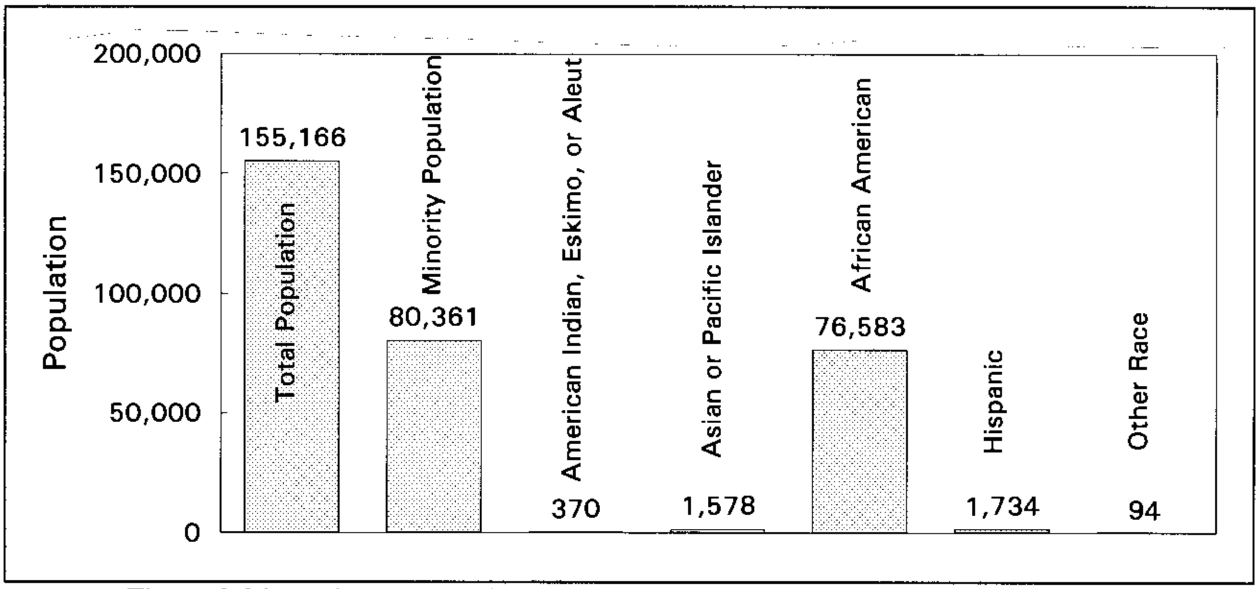


Figure 3-34 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Savannah

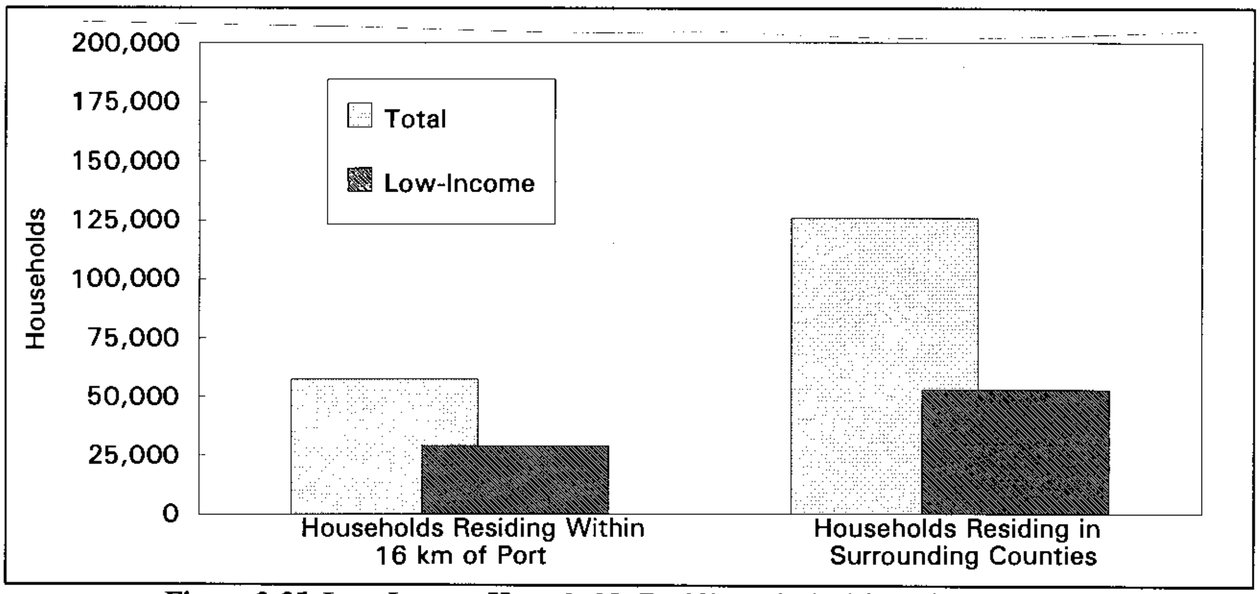


Figure 3-35 Low-Income Households Residing within 16 km (10 mi) of the Port of Savannah

3.2.1.9 Tacoma, WA

The Port of Tacoma is located in the southeastern corner of Puget Sound on the deep waters of Commencement Bay about 5 km (3 mi) from the Sound. It is a rapidly expanding major port second only to Seattle in maritime importance on Puget Sound. The distance from the entrance into Puget Sound is approximately 130 km (80 mi). A map of the port is shown in Figure 3-36.

Terminal 7, Berth D is the primary container terminal. It has one 274 m (904 ft) long container berth, 3 container cranes, and 15.2 m (50 ft) of depth alongside at mean low water.

The terminal is about 4.8 km (3 mi) from the Port of Tacoma road access to Interstate 5 immediately outside the port complex. A somewhat longer route, Interstate 5 South, connects with I-84 East near Portland, OR. Ship berths are served by the Port Belt Line Railroad, and the port is served by the Burlington Northern and Union Pacific Railroads, which interline with eastern and southern railroads.

The 1990 population within 16 km (10 mi) of the port was 511,575. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 601,000; the Oak Ridge Reservation, 431,000; the Idaho National Engineering Laboratory, 157,000; the Hanford Site, 98,600; and the Nevada Test Site, 379,000. Populations along rail routes to four of these sites are slightly larger, but the population along the rail route to the Nevada Test Site is slightly smaller (this is largely due to primary use of interstate highways through Salt Lake City, UT and Las Vegas, NV). The distances to the five potential sites on interstate routes are: the Savannah River Site, 4,720 km (2,933 mi); the Oak Ridge Reservation, 4,280 km (2,659 mi); the Idaho National Engineering Laboratory, 1,310 km (814 mi); the Hanford Site, 399 km (248 mi); and the Nevada Test Site, 2,160 km (1,342 mi). Distances along rail routes are much longer.

Environmental Conditions: A variety of marine mammals can be found in central Puget Sound including the Pacific harbor seal, California sea lion, killer whale, Dall porpoise, and harbor porpoise. In 1991, the U.S. National Marine Fisheries Services reported that the following endangered and/or threatened species may occur in the Puget Sound: the gray whale, the humpback whale, the Stellar sea lion, and the endangered leatherback sea turtle (DOE, 1995c), although these species are not reported at the port. Bald eagles can be found throughout this coastal zone, and American peregrine falcons are uncommon winter visitors (FWS, 1981a). The U.S. Fish and Wildlife Services' Ecological Inventory for the Puget Sound area indicates that the habitat of Commencement Bay is used by a variety of birds including shorebirds, gulls, sandpipers, turnstones, yellowlegs, herons, rails, great blue herons, waterfowl, loons, grebes, swans, geese, dabbling ducks, diving ducks, mergansers, American wigeons, pintails, mallards, seabirds, cormorants, alcids, common murres, and pigeon guillemots. Adult concentrations of all of these species may be found in the Bay. Some of these species may also use this area as an overwintering area, a migratory area, and/or a nesting area (FWS, 1981a). It is also indicated that adult concentrations of chinook salmon, coho salmon, chum salmon, and pink salmon are found in the Puyallup Waterway/River and use this water body and upstream segments as migratory and nursery areas.

According to the State of Washington's Department of Wildlife, a number of seabird colonies exist along the shoreline of Commencement Bay. Areas of the Puget Sound, north of Commencement Bay, are also used as haulouts by the California Sea Lion. Areas of estuarine wetlands are located along the northern shore of Commencement Bay (WDW, 1994).

Climatic Conditions: The mild climate of the Pacific Coast is modified by the Cascade Mountains and to a lesser extent by the Olympic Mountains. The climate is characterized by mild temperatures, a well-defined rainy season and prolonged cloud cover, especially during the winter months. The Cascades act as a very effective barrier in both winter and summer, shielding the region from both extreme cold and

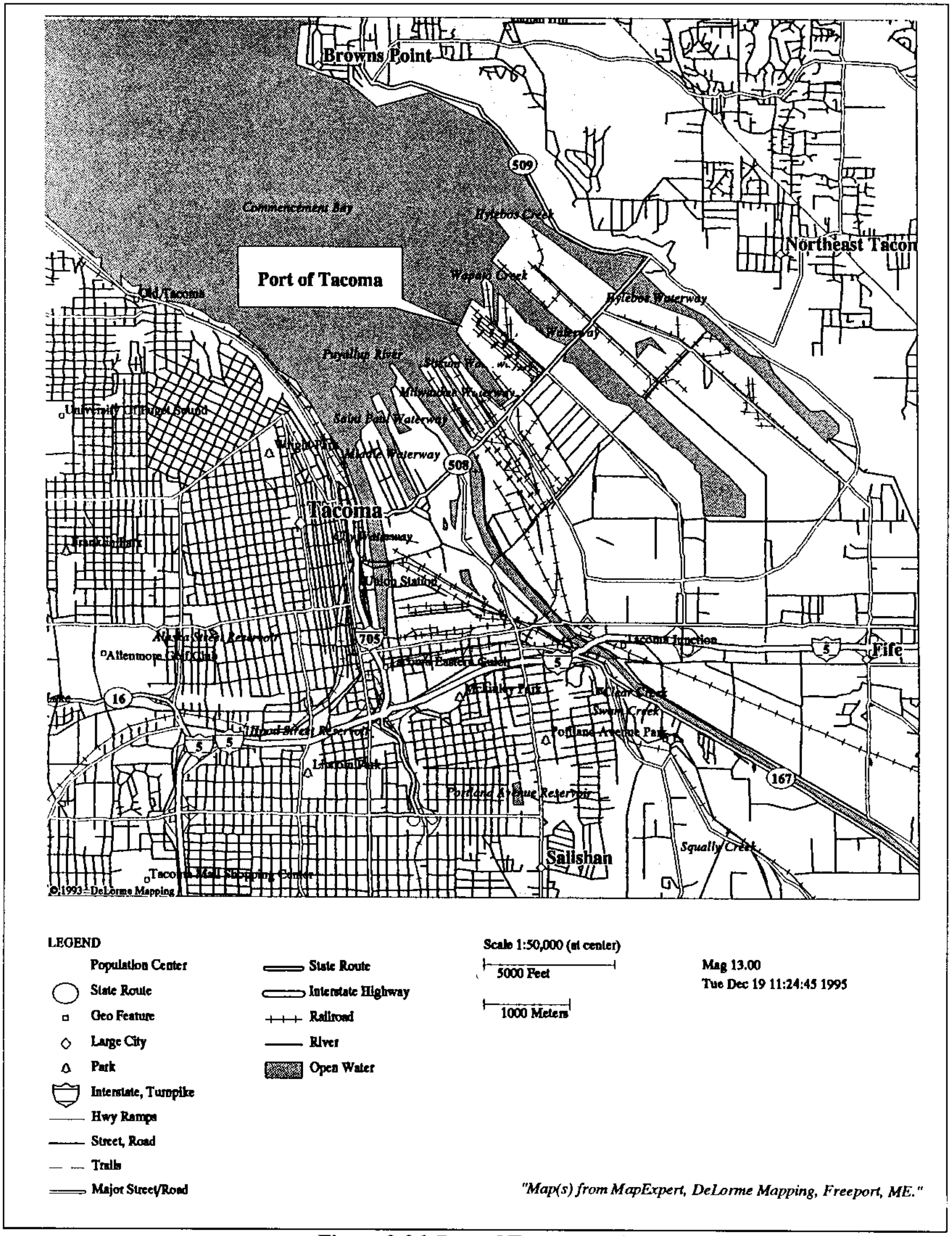


Figure 3-36 Port of Tacoma, WA

heat, respectively. The rainy season extends from October through March, with December accounting for the most rainfall. Approximately 75 percent of the annual total precipitation occurs during the winter rainy season. The dry season is centered around July and August. The majority of Seattle's precipitation is associated with normal, mid-latitude disturbances, which are most vigorous during the winter months. During summer, the dominant storm track (e.g., the polar jet) shifts northward into southern Canada, reducing the precipitation in the area. Summer thunderstorms do occur but do not contribute measurably to the annual rainfall budget. Prevailing winds are from the southwest, but occasional severe winter storms will produce strong northerly winds. Summer winds are generally rather light, with the occasional evidence of land-sea breeze effects creating northerly flows. Fog and low-level stratocumulus clouds form over the southern Puget Sound area in the late summer, fall, and early winter months, and often dominate the weather conditions of the early morning hours, reducing surface visibility. Based on 1951-1980 climatology, the first occurrence of freezing temperatures should occur around November 11, and the last incidence in spring around March 24 (NOAA, 1992f).

The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code. For the Port of Tacoma, the Uniform Building Code provides a basic wind speed of about 130 km per hour (80 mph) (UBC, 1991). The port is located in a high seismic zone with an acceleration of 0.30 g. There have been two major earthquakes in the Puget Sound area this century (Bolt, 1978). On May 18, 1980, Mount St. Helens suffered a major volcanic eruption (IPA, 1993). All the mountains along the Cascade Range from Canada to Northern California are volcanic in origin and are potentially active.

Ethnic and Income Characteristics: Figure 3-37 shows the ethnic composition for the area surrounding the port at Tacoma, WA. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans constituted the largest minority group at about 6 percent of the total population, and approximately 38 percent of the minority population for the area surrounding the port. Asian and Hispanic minorities make up approximately 33 percent and 20 percent, respectively, of the minority population. Native Americans make up about 8 percent of the minority population near the port of Tacoma. Figure 3-38 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.10 Wilmington, NC

The Port of Wilmington, NC, is located on the east bank of the Cape Fear River, about 42 km (26 mi) above its mouth. It is the leading port of North Carolina, and its major export is wood pulp. The major terminals are down river from the city. A Federal project maintains a 12.2 m (40 ft) channel at the mouth of the Cape Fear River, 11.6 m (38 ft) to the port. A new dredging program will deepen the approach channel to 12.2 m (40 ft). A map of the port is shown in Figure 3-39.

The Wilmington wharves are of concrete pile construction, rubber fendered, with a total frontage of about 2,000 m (6,600 ft). Berths 6 to 9 are dedicated containership berths, with the remaining berths used for various kinds of general cargo. All of the main cargo berths have a depth alongside of 11.6 m (38 ft) at mean low water. The terminal has five container cranes, plus three gantry cranes (Jane's, 1992; AAPA, 1993; FHI, 1993).

Truck shipments from the port to southern destinations are along U.S. Routes 17, 74, 76, and 421 to Interstates 95 and 40 (POW, 1994). Northern and western long-distance routes are via Interstate 40, which connects with State Highway 132 about 16 km (10 mi) north of the city. The 1990 population within

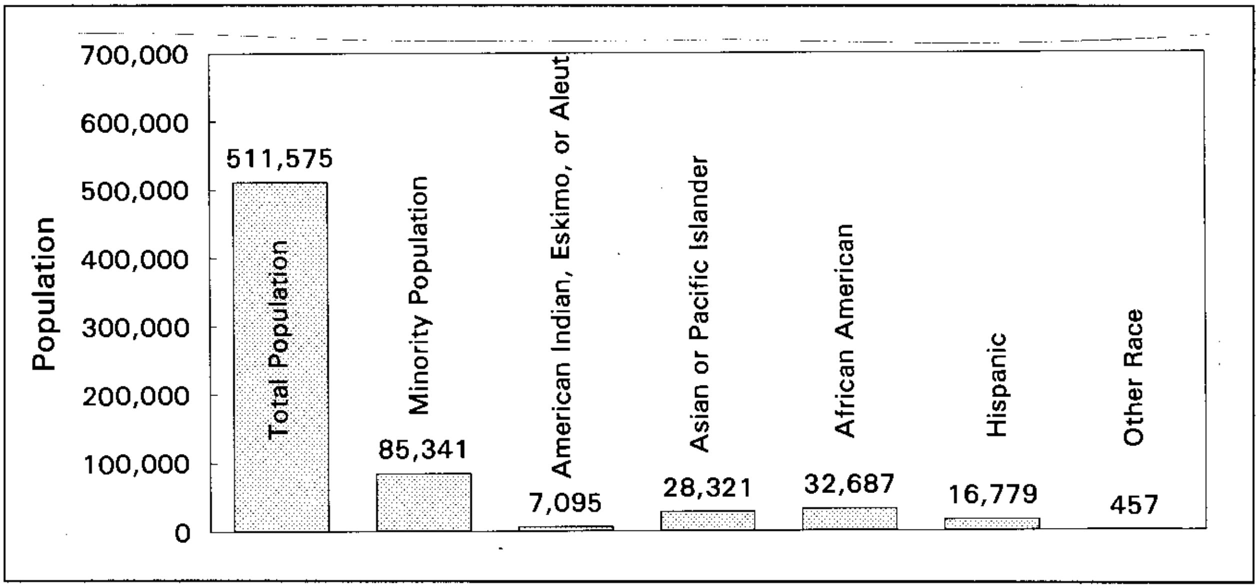


Figure 3-37 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Tacoma

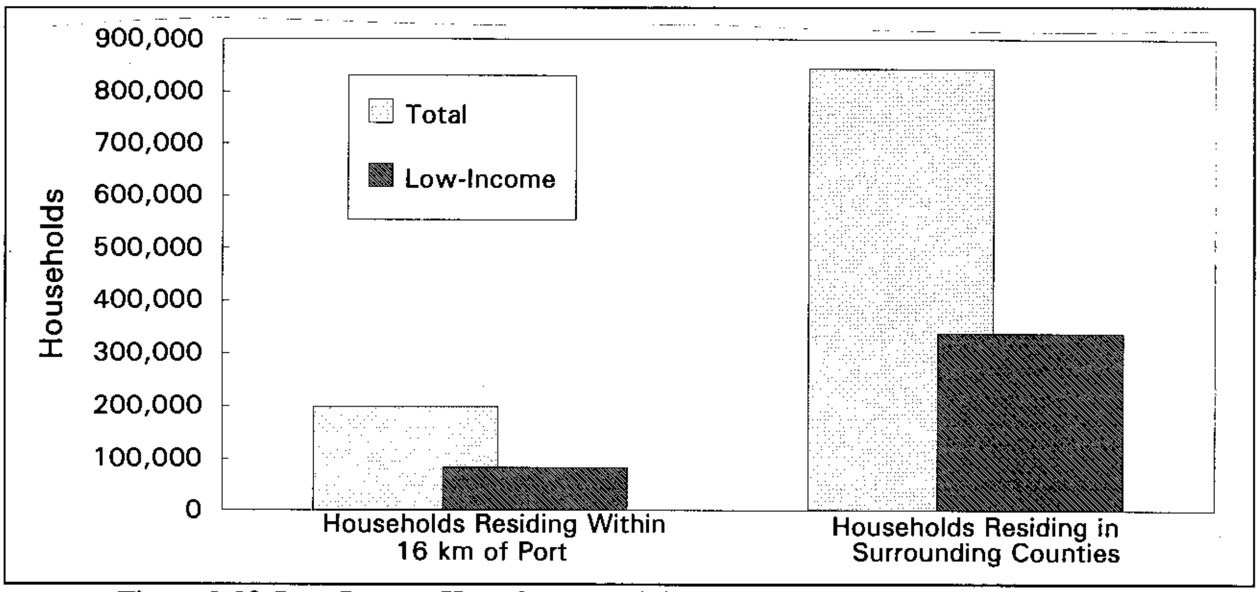


Figure 3-38 Low-Income Households Residing within 16 km (10 mi) of the Port of Tacoma

16 km (10 mi) of the port terminals was 115,057. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 64,700; the Oak Ridge Reservation, 128,000; the Idaho National Engineering Laboratory, 507,000; the Hanford Site, 556,000; and the Nevada Test Site, 570,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 500 km (311 mi); the Oak Ridge Reservation, 820 km (510 mi); the Idaho National Engineering Laboratory, 4,100 km (2,548 mi); the Hanford Site, 4,770 km (2,964 mi); and the Nevada Test Site, 4,260 km (2,647 mi). Distances along rail routes are slightly longer for western sites, but about the same for eastern sites.

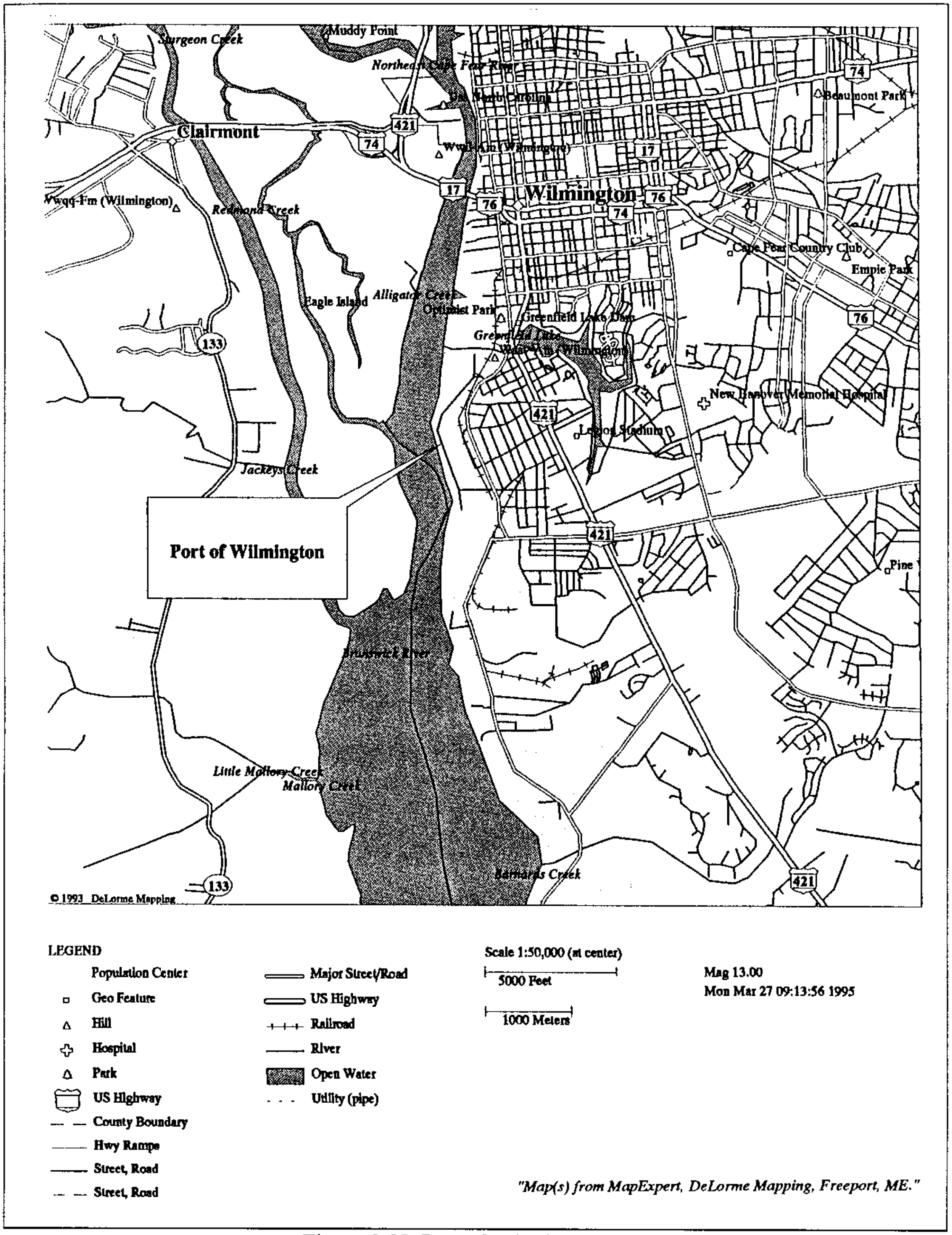


Figure 3-39 Port of Wilmington, NC

Environmental Conditions: There are no known environmentally sensitive areas in the immediate vicinity of the terminal, but due to resorts and recreational activity, there is heightened environmental awareness.

North Carolina has given the lower portion of the Cape Fear River three different stream classifications. From the Northeast Cape Fear River to the confluence with the Cape Fear River the waters are classified as SC-swamp. From the mouth of the Northeast Cape Fear to a point between Snow and Federal Points, the waters are classified as SC, and from Snow and Federal Points oceanward the waters are classified as SA. SC waters are tidal waters suitable for fishing, fish and wildlife propagation, secondary recreation, and other water uses requiring lower quality. The term "swamp" denotes waters with slow velocity. Class SA waters are suitable for shellfishing and primary recreation, as well as all of the activities approved for Class SC waters. According to the U.S. Fish and Wildlife Service's Ecological Inventory Map for Beaufort, NC, the Port of Wilmington is located in a low salinity estuarine habitat (generally 0.5 to 5 ppt) and tidal freshwater habitat. Below Wilmington at Campbell Island, the river changes to a mid-salinity estuarine habitat (generally 5 to 16.5 ppt). The Cape Fear River near MOTSU changes once again to a high-salinity estuarine habitat (generally 16.5 to 30 ppt).

The lower Cape Fear River supports a large number of aquatic and terrestrial species. There are both invertebrate and fish species of commercial and recreational value found in the Cape Fear River near the Port of Wilmington. Species sought by commercial and recreational fishermen include flounder, trout, spot, croaker, bluefish, Spanish mackerel, and king mackerel. Shellfish sought include penaeid shrimp and blue crabs.

The Natural Heritage Program of the North Carolina Department of Environment, Health and Natural Resources reports that the area around the port has not been systematically inventoried for rare species. However, they also report that the lower Cape Fear River, from Wilmington to the mouth of the river at Smith Island, is brackish and contains numerous rare animals. The shortnose sturgeon (State and Federal Endangered Species) rarely occurs in the river, whereas manatees (State and Federal Endangered Species) occasionally occur, especially in the summer. American alligators (a designated threatened species) can be found in tributary streams. The freckled blenny, spinycheek sleeper, opossum pipefish, and marked goby are other rare marine fishes that inhabit the river.

There are many animals with special status in this area including various types of whales, sea turtles, and birds. State or Federally protected, endangered, or threatened aquatic species in this area include the shortnose sturgeon (fish), finback whale, Florida manatee, humpback whale, right whale, sei whale, and sperm whale (mammals), Arctic peregrine falcon, bald eagle, piping plover, red-cockaded woodpecker, wood stork (birds), and the American alligator, green sea turtle, hawksbill sea turtle, Kemp's ridley sea turtle, leatherback sea turtle, and the loggerhead sea turtle (reptiles and amphibians).

Climatic Conditions: The Port of Wilmington is located on the Cape Fear River, 42 km (26 mi) from the open Atlantic Ocean. This general area also includes MOTSU, which is also located on the Cape Fear River, north of Southport, NC, and south of Wilmington, NC. The elevation of this region is approximately 12 m (40 ft) above sea level, and is more variable than the coastal plain surrounding the Norfolk, VA area. The National Weather Service has been archiving meteorological information for this area since 1871.

The port is subject to hurricanes and tropical storms. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code. For the Port of Wilmington, the Uniform Building Code provides a basic wind speed of about 160 km per hour (100 mph) (UBC, 1991). The port is located in a low seismic zone with an acceleration of 0.075 g.

The maritime location of the Wilmington area makes the climate unusually mild for its northern latitude. All wind directions from the east-northeast through the southwest have some moderating effect on the local climate, due to the relatively warm and cool ocean in the winter and summer seasons, respectively. The area rarely experiences cold episodes where the temperature falls below -18°C (0°F). However, cold air outbreaks do occur, causing sharp fluctuations in winter temperatures. Rainfall in the area is generally considered ample and evenly distributed throughout the year, with the bulk of the precipitation occurring during the summer months. The bulk of this rainfall is generally associated with afternoon and evening thunderstorms. In contrast, the winter rains tend to be of the slow, steady type, generally lasting one to two days. As is common at Atlantic coastal localities at this latitude, the late summer and early fall months bring the possibility of hurricanes and tropical storms to the Wilmington area. These storms are capable of generating high winds, above normal tides and torrential rains. The latter two are also capable of creating widespread local flooding of low-lying coastal areas (NOAA, 1992g).

Ethnic and Income Characteristics: Figure 3-40 shows the ethnic composition for the area surrounding the port at Wilmington. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 33 percent of the total population, and approximately 93 percent of the minority population for the area surrounding the port. Figure 3-41 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

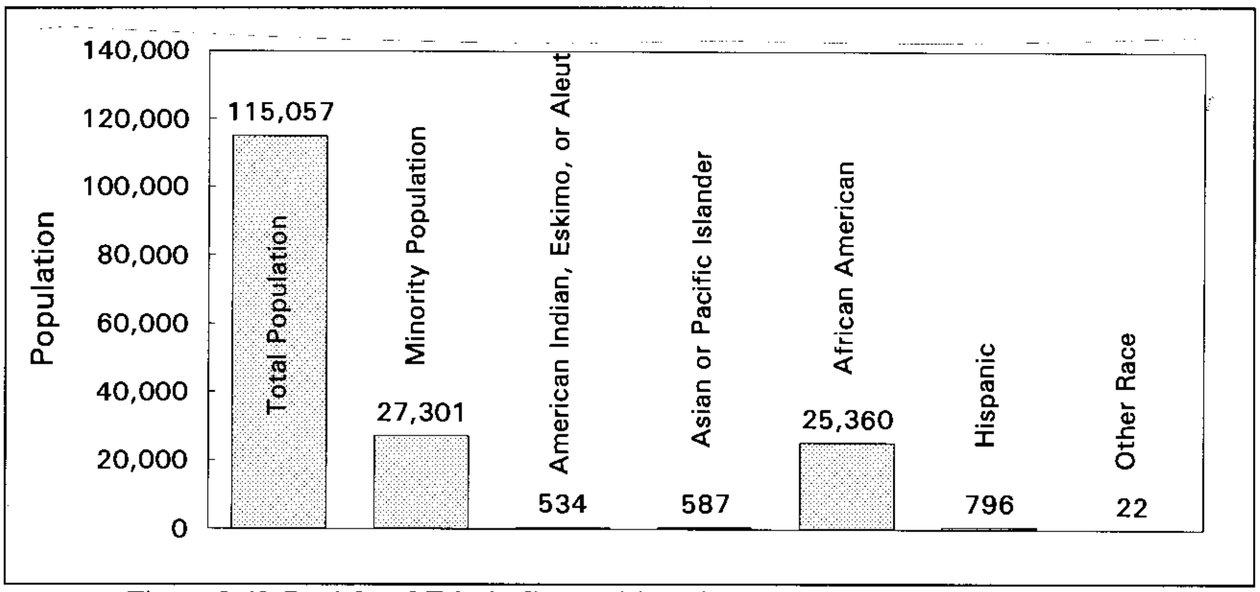


Figure 3-40 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Wilmington

3.3 Management Site(s) Environments

This section describes the affected environment of the five potential DOE management sites for the foreign research reactor spent nuclear fuel. The five management sites are the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

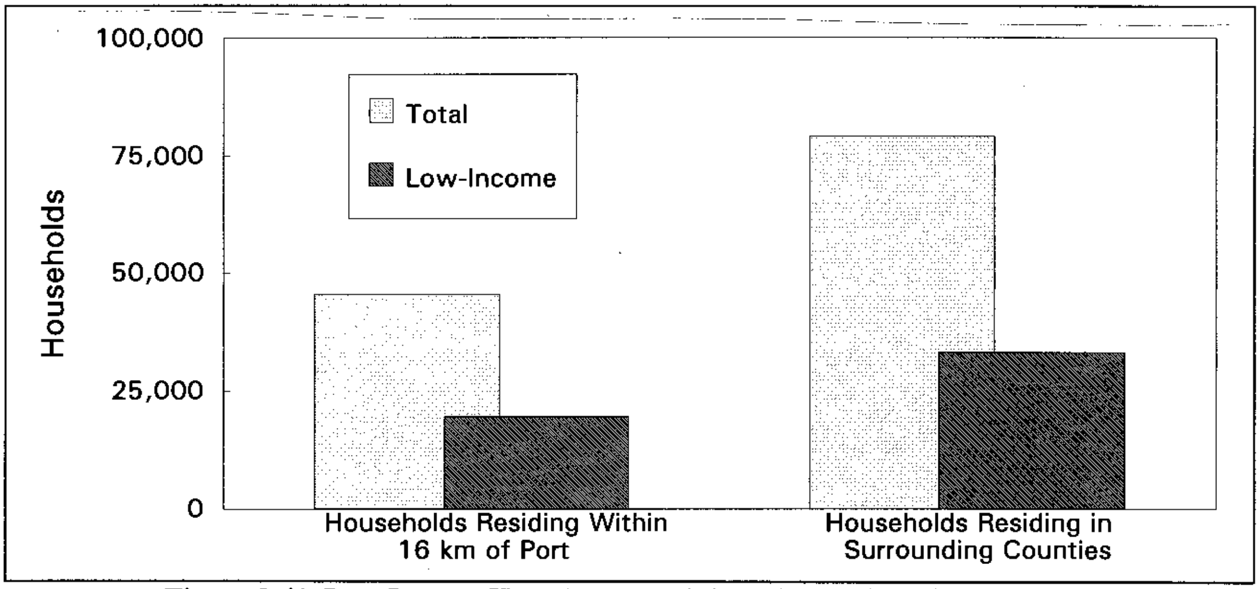


Figure 3-41 Low-Income Households Residing within 16 km (10 mi) of the Port of Wilmington

3.3.1 Description of the Affected Environment at the Savannah River Site

The Savannah River Site is a key DOE facility for research and production of special nuclear materials. The site was built in the early 1950's to produce the basic materials used in the fabrication of nuclear weapons. The DOE Savannah River Operations Office manages the Savannah River Site, and the Westinghouse Savannah River Company operates the site under contract to DOE. This section describes the potentially affected environment of the Savannah River Site. The location of the site is shown in Figure 3-42.

3.3.1.1 Geology

The Savannah River Site is located in the Upper Atlantic Coastal Plain physiographic province of western South Carolina, approximately 32 km (20 mi) southeast of the Fall Line, which separates the Piedmont and Coastal Plain provinces (Figure 3-42). The Coastal Plain in South Carolina is subdivided to include the Aiken Plateau, the Congaree Sand Hills, and the Coastal Terraces. The Coastal Plain consists of 213 to 366 m (700 to 1,200 ft) of gently seaward (southeast) dipping sands, clays, and limestones of Cretaceous and Tertiary age. These sediments are underlain by sandstones of Triassic age and older dense metamorphic and igneous basement rocks (Arnett et al., 1993). Coastal Plain sediments form a wedge of seaward-dipping and thickening unconsolidated and semi-consolidated sediments that begin at zero at the Fall Line and increase to more than 1,212 m (4000 ft) at the Continental Shelf. The Coastal Plain sediments underlying the Savannah River Site consist of sandy clays and clayey sands, with occasional beds of clean sand, gravel, clay, or carbonate. Two bioclastic limestone zones ranging from 0.6 m (2 ft) to 24 m (80 ft) occur within the Tertiary sequence. Most of the clastic sediments are unconsolidated, but thin semi-consolidated beds also occur (DOE, 1991a). The Triassic formations and older igneous and metamorphic rocks are hydrologically separated from the overlying Coastal Plain sediments by a regional aquitard (Arnett et al., 1993) (Figure 3-43).

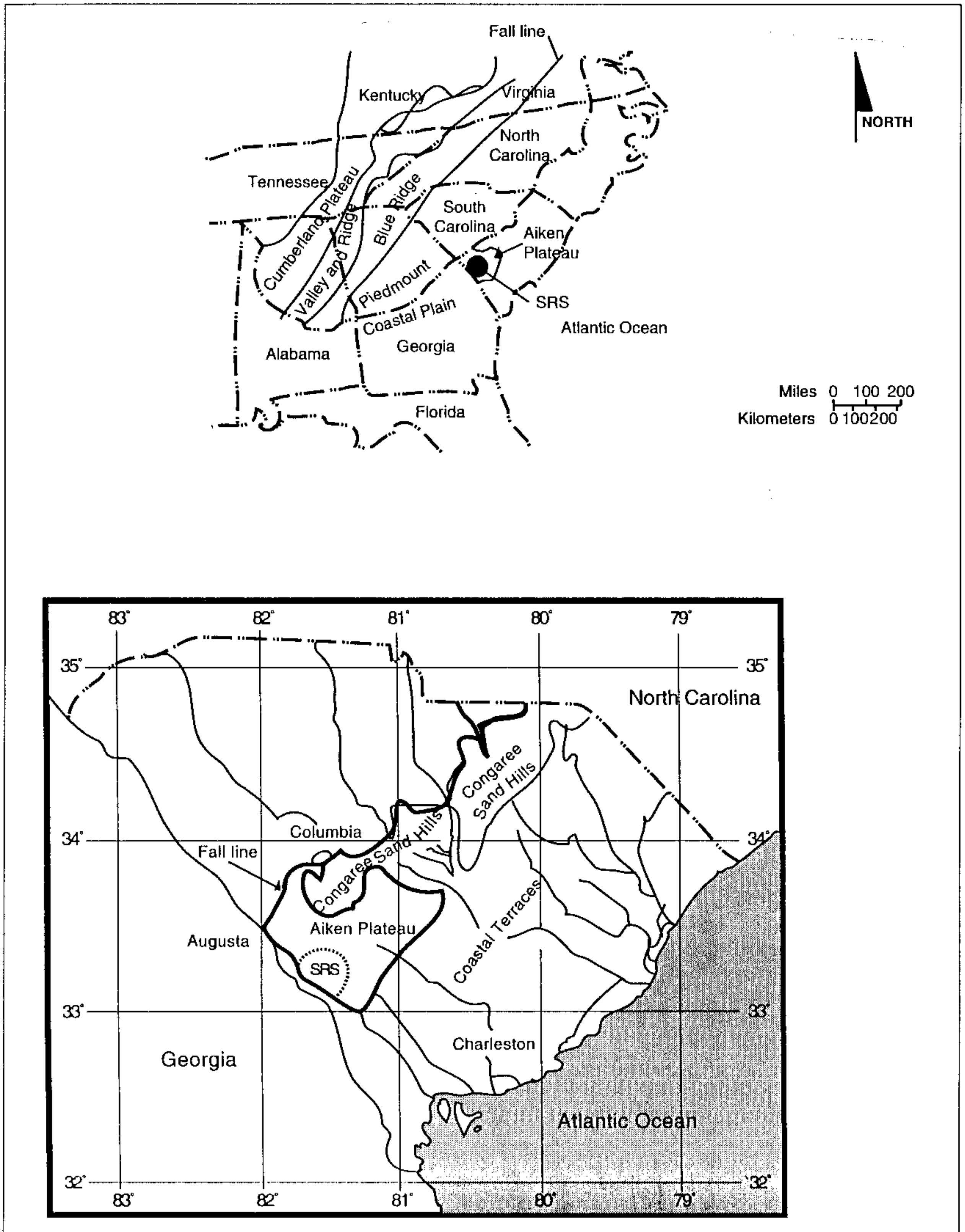


Figure 3-42 Location of the Savannah River Site in the Southern United States

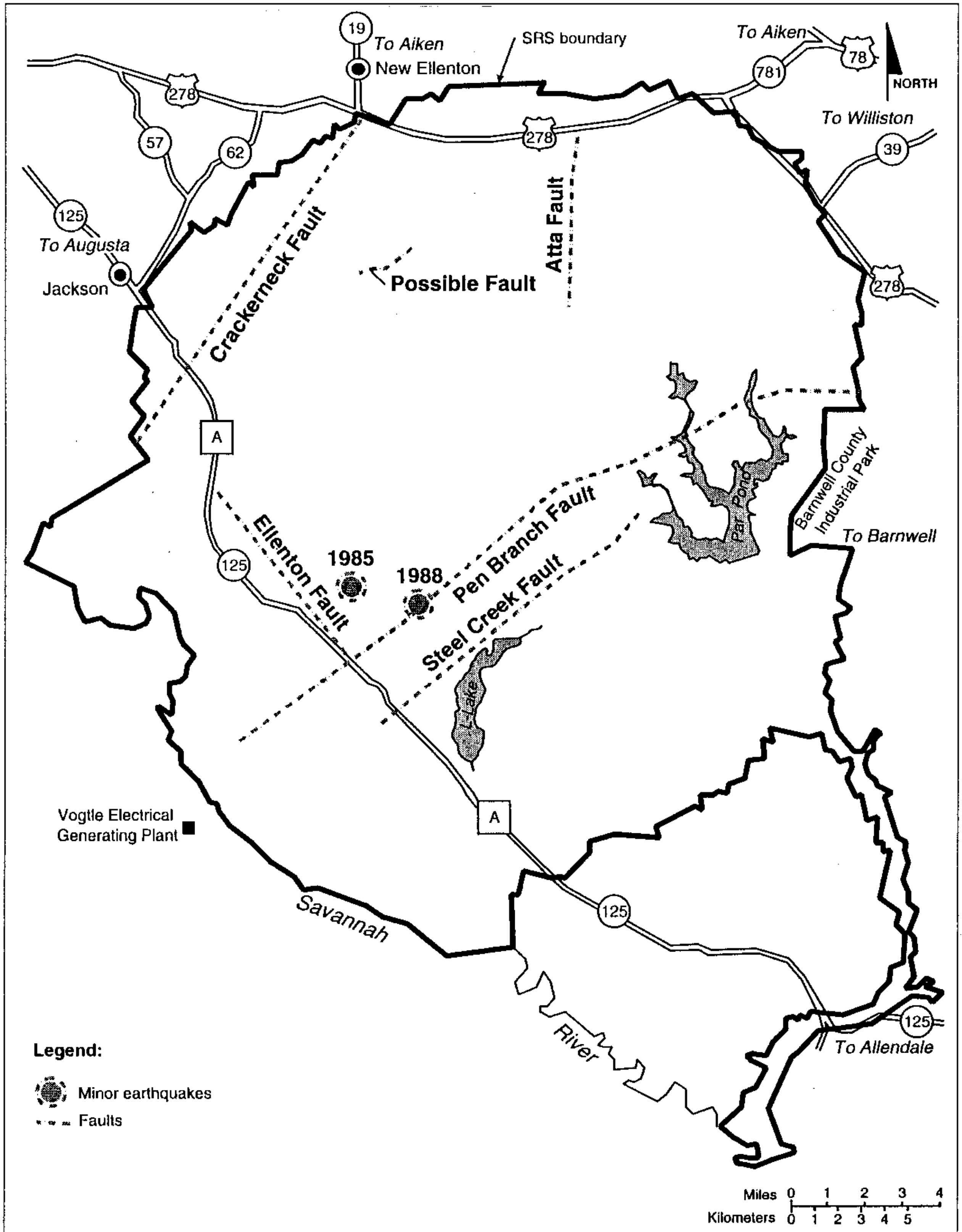


Figure 3-43 Geologic Faults of the Savannah River Site

3.3.1.2 Seismology and Volcanology

Seismicity in the Coastal Plain of South Carolina occurs in three distinct seismic zones near the Charleston area: Middleton Place Summerville, about 19 km (12 mi) northwest of Charleston; Bowman, about 59 km (37 mi) northwest of the Middleton Place-Summerville; and Adams Run, about 30 km (19 mi) southwest of the Middleton Place-Summerville (WSRC, 1993a). Of the three 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 Savannah River Site, both in terms of maximum historic site intensity and the number of earthquakes felt in the area (WSRC, 1993a).

The closest offsite fault system is the Augusta Fault Zone, approximately 40 km (25 mi) from the Savannah River Site. In this fault zone, the Belair Fault has experienced the most recent movement, but is not considered capable of generating major earthquakes (DOE, 1987). There is no conclusive evidence of recent displacement along any fault within 320 km (200 mi) of the Savannah River Site, with the possible exception of the buried faults in the epicentral area of the 1886 Charleston, SC earthquake, approximately 144 km (90 mi) away (DOE, 1991a).

Two notable earthquakes have occurred within 320 km (200 mi) of the Savannah River Site. The first was a major earthquake in 1886 centered in the Charleston area, which had an estimated Richter magnitude of 6.8. The second earthquake was the Union County, SC earthquake of 1913, which had an estimated Richter magnitude of 6.0, and occurred about 160 km (100 mi) from the Savannah River Site (WSRC, 1993a).

Two earthquakes have occurred at the Savannah River Site during recent years. In June 1985, onsite instruments recorded an earthquake with a magnitude of 2.6 and a focal depth of about 1.0 km (0.6 mi) (DOE, 1995c). The epicenter was just west of the C- and K-areas. In August 1988, an earthquake of magnitude 2.0 and a focal depth of approximately 2.7 km (1.7 mi) occurred (Stephenson, 1988).

3.3.1.3 Hydrology

3.3.1.3.1 Surface Water

The Savannah River bounds the Savannah River Site on its southwestern border for about 32 km (20 mi), approximately 260 river km (160 river mi) from the Atlantic Ocean. At the Savannah River Site, the Savannah River flow averages about 283 m^3 per sec (74,760 gal per sec). Five principal tributaries to the Savannah River are found on the Savannah River Site: Upper Three Runs Creek, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs Creek (Figure 3-44). These tributaries drain almost all of the Savannah River Site. Each of these streams originates on the Aiken Plateau in the Coastal Plain, and descends 15 to 60 m (50 to 200 ft) before discharging into the river. The streams, which historically have received varying amounts of discharge from the Savannah River Site operations, are not commercial sources of water. The natural flow of the Savannah River Site streams ranges from less than 1 m^3 per sec (264 gal per sec) in smaller streams such as Pen Branch to 6.8 m^3 per sec (1,795 gal per sec) in Upper Three Runs. Three large upstream reservoirs - 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 Savannah River.

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 Savannah River Site, the river supplies domestic and industrial water needs for Augusta, GA, and North Augusta, SC. Downstream

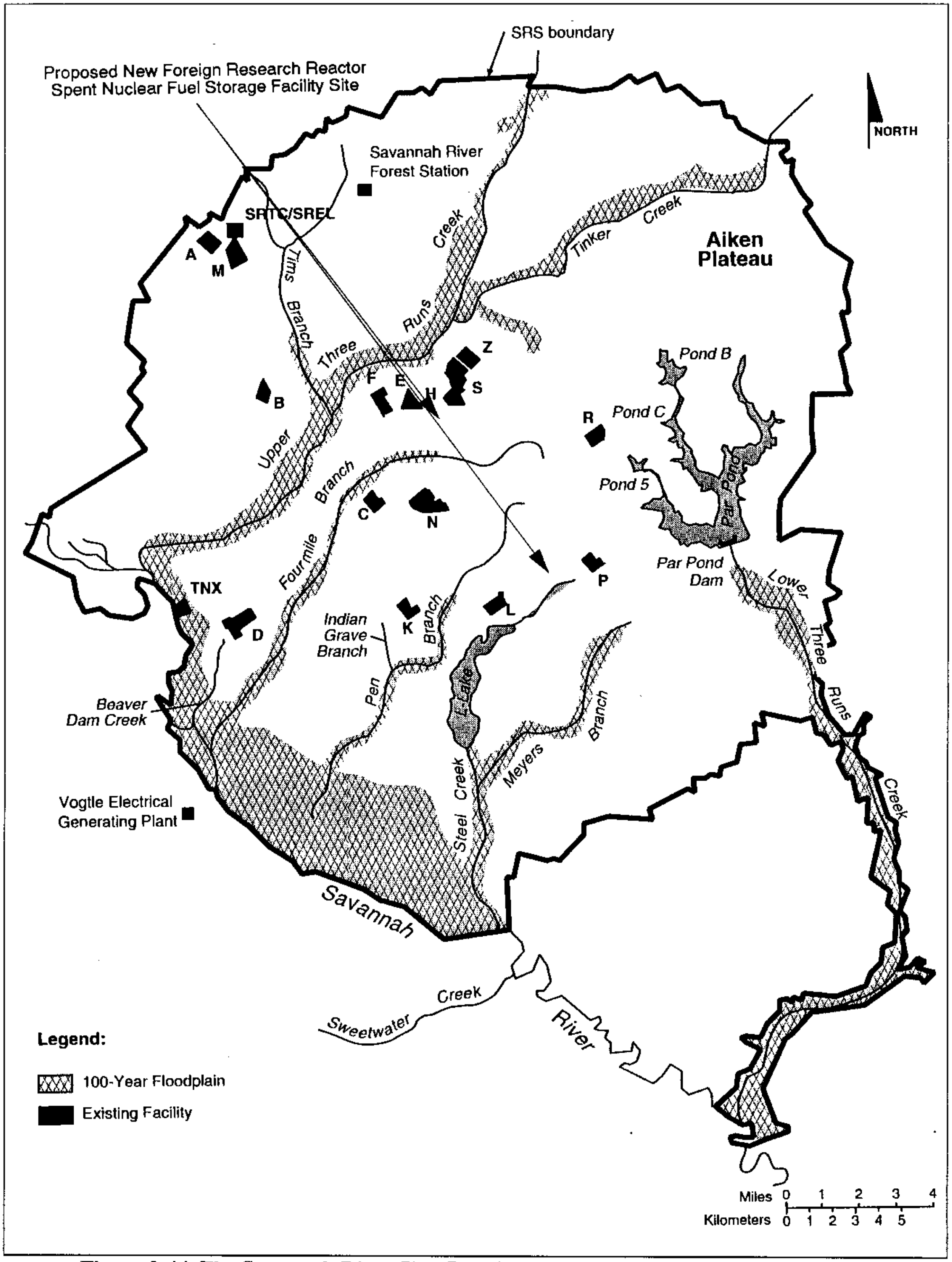


Figure 3-44 The Savannah River Site, Showing 100-Year Floodplain, Major Stream Systems and Facilities

of the Savannah River Site, the river supplies domestic and industrial water needs for Savannah, GA, and Beaufort and Jasper Counties in South Carolina. The South Carolina Department of Health and Environmental Control regulates the physical properties and concentrations of chemicals and metals in the Savannah River Site effluent under the National Pollutant Discharge Elimination System. This department also regulates chemical and biological water quality standards for the Savannah River Site waters. On April 24, 1992, the department changed the classification of the Savannah River and the Savannah River Site 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 of Appendix C, Volume 1 of the Programmatic SNF&INEL Final EIS list the characteristics of the Savannah River Site surface water quality (DOE, 1995c).

3.3.1.3.2 Groundwater

There are two hydrogeologic provinces in the subsurface beneath the Savannah River Site. The deepest Piedmont hydrogeologic province includes Paleozoic metamorphic and igneous basement rocks, and Triassic-aged lithified mudstone and sandstone. The Southeastern Coastal Plain hydrogeologic province lies above the Piedmont province and consists of a seaward thickening wedge of unconsolidated sediments of Late Cretaceous and Tertiary age. The Southeastern Coastal Plain hydrogeologic province is more important from a resource point of view because it holds an abundant supply of high-quality groundwater.

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. The Appleton system separates the Southeastern Plain hydrogeologic province from the underlying Piedmont hydrogeologic province. Locally, individual aquifer and confining units are delineated. The complexly interbedded strata that form the three aquifer systems primarily consist of fine-to-coarse-grained sand and local gravel and limestone deposited under relatively high energy conditions in fluvial to shallow marine environments.

The water table receives water through rainfall percolating through the vadose zone. The deeper semi-confined aquifers receive water from groundwater flow into the Savannah River Site from offsite or from water flowing from aquifers above or below. The direction of groundwater flow in the vadose zone is predominantly downward, but some lateral flow occurs because of clay lenses in the soil. The flow of groundwater in the water table and deeper semi-confined aquifers is controlled by the hydraulic properties of the sediments (e.g., conductivity) and the proximity to streams. Savannah River ultimately receives all groundwater that flows beneath the Savannah River Site, and no contaminated groundwater is flowing off of the Savannah River Site.

Groundwater Quality: The quality of groundwater in the principal hydrologic systems beneath the Savannah River Site 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, 1993b). In general, the quality of the groundwater in the Coastal Plain sediments at the Savannah River Site and the surrounding areas is suitable for most domestic and industrial purposes. The waters are dilute with respect to total dissolved solids concentrations, which range from less than 10 mg per L to about 150 to 200 mg per L. The pH values range from as low as 4.9 to a maximum value of 7.7 (where the groundwater is in contact with limestone). Due to the low solids content of the waters and the frequently low pH values, many of the waters are corrosive to metal surfaces. High dissolved iron concentrations can also be of concern in some units. An onsite degasification and filtration process raises the pH and removes iron in domestic water supplies where necessary (WSRC, 1993b). Table 4-12 of Appendix C, Volume 1 of the Programmatic SNF&INEL Final EIS summarizes the Savannah River Site groundwater quality data, and Table 4-13 lists data for radiological constituents (DOE, 1995c).

The groundwater beneath 5 to 10 percent of the Savannah River Site has been contaminated by industrial solvents, metals, tritium, or other constituents used or generated on the site. Figure 3-45 shows the locations of facilities monitored by the Savannah River Site and areas with constituents that exceeded drinking water standards in 1992. In general, contaminated groundwater at the Savannah River Site is beneath a few facilities, and the contaminants reflect operations and chemical processes at those facilities. 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 found at A- and M-Areas), and sulfate. The groundwater beneath the Sanitary Landfill contains chlorinated volatile organics, radionuclides, and metals. The groundwater beneath all the reactor areas except R-Area contains tritium, other nuclides, metals, and chlorinated volatile organics, and 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).

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 Gordan aquifer supplies some municipal, industrial, and agricultural users. The Gordan and Upper Three Runs Creek aquifers are the primary sources of domestic water supplies in the vicinity of the Savannah River Site for rural non-municipal water. DOE has identified 56 major municipal, industrial, and agricultural groundwater users within 32 km (20 mi) of the center of the Savannah River Site (DOE, 1987). The total pumpage for these users is about 136,260 m^3 per day (36 million gal per day).

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 (DOE, 1995c), or U.S. Environmental Protection Agency Class II, meaning that the aquifers can provide resource-quality water, but are not the sole source of supply (as are South Carolina Class GA or U.S. Environmental Protection Agency Class 1 aquifers) (DOE, 1991a).

3.3.1.4 Meteorology

Wind: Figure 3-46 shows annual wind direction frequencies and wind speeds for the Savannah River Site from 1987 through 1991. The maximum wind directional frequencies are from the northeast and west-southwest. The average wind speed for this 5-yr period was 3.8 m per sec (8.5 mph). Calm winds (less than 2 m per sec or 4.5 mph) occurred less than 10 percent of the time during the 5-yr period. Seasonally, wind speeds were greatest during the winter at 4.1 m per sec (9.2 mph), and lowest during the summer at 3.4 m per sec (7.6 mph) (Shedrow, 1993). Winter snow storms in the Savannah River Site area occasionally bring strong and gusty surface winds with speeds as high as 32 m per sec (72 mph). Thunderstorms can generate winds with speeds as high as 18 m per sec (40 mph) and even stronger gusts. The fastest wind speed recorded at Augusta between 1950 and 1986 was 37 m per sec (83 mph) (DOE, 1995c).

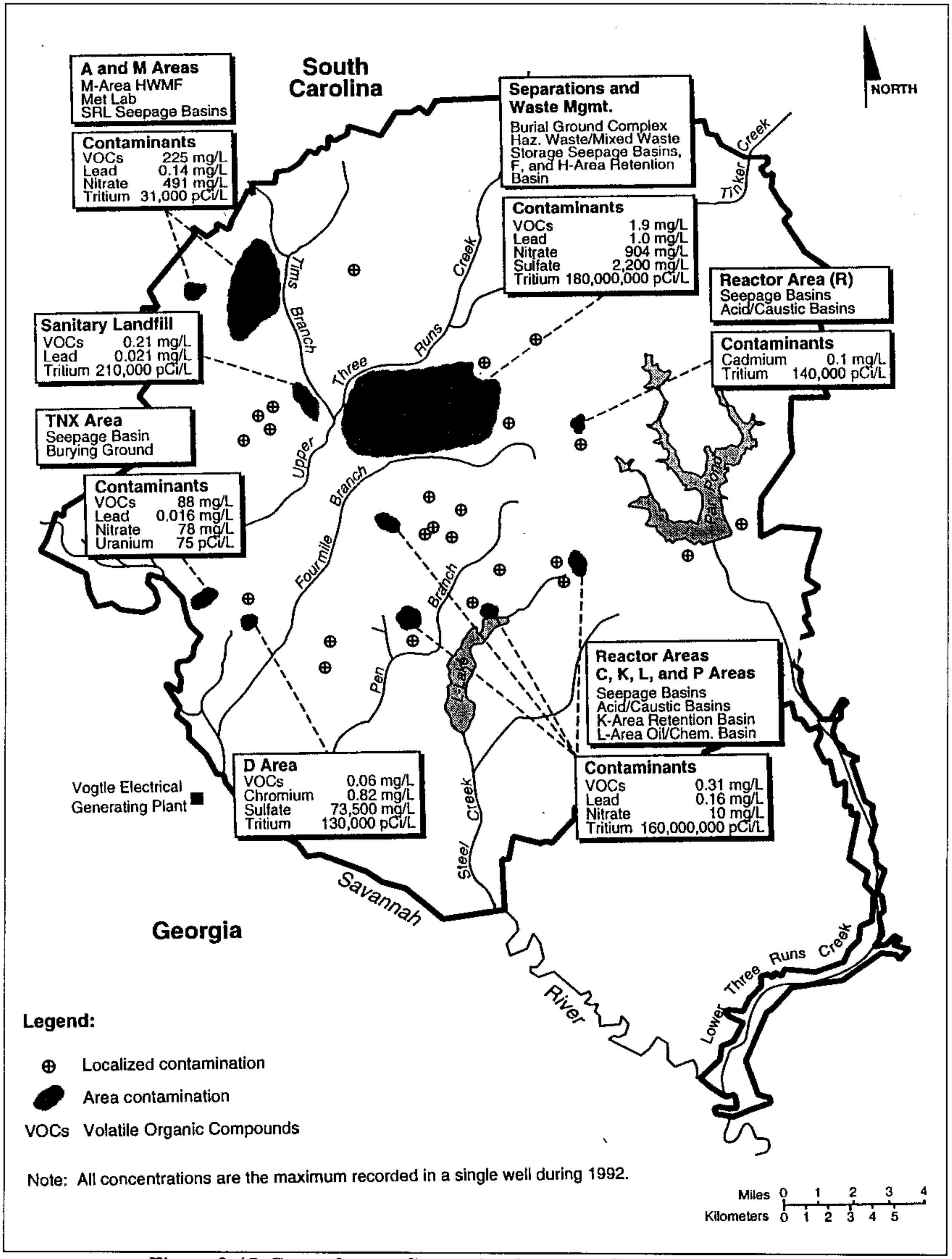
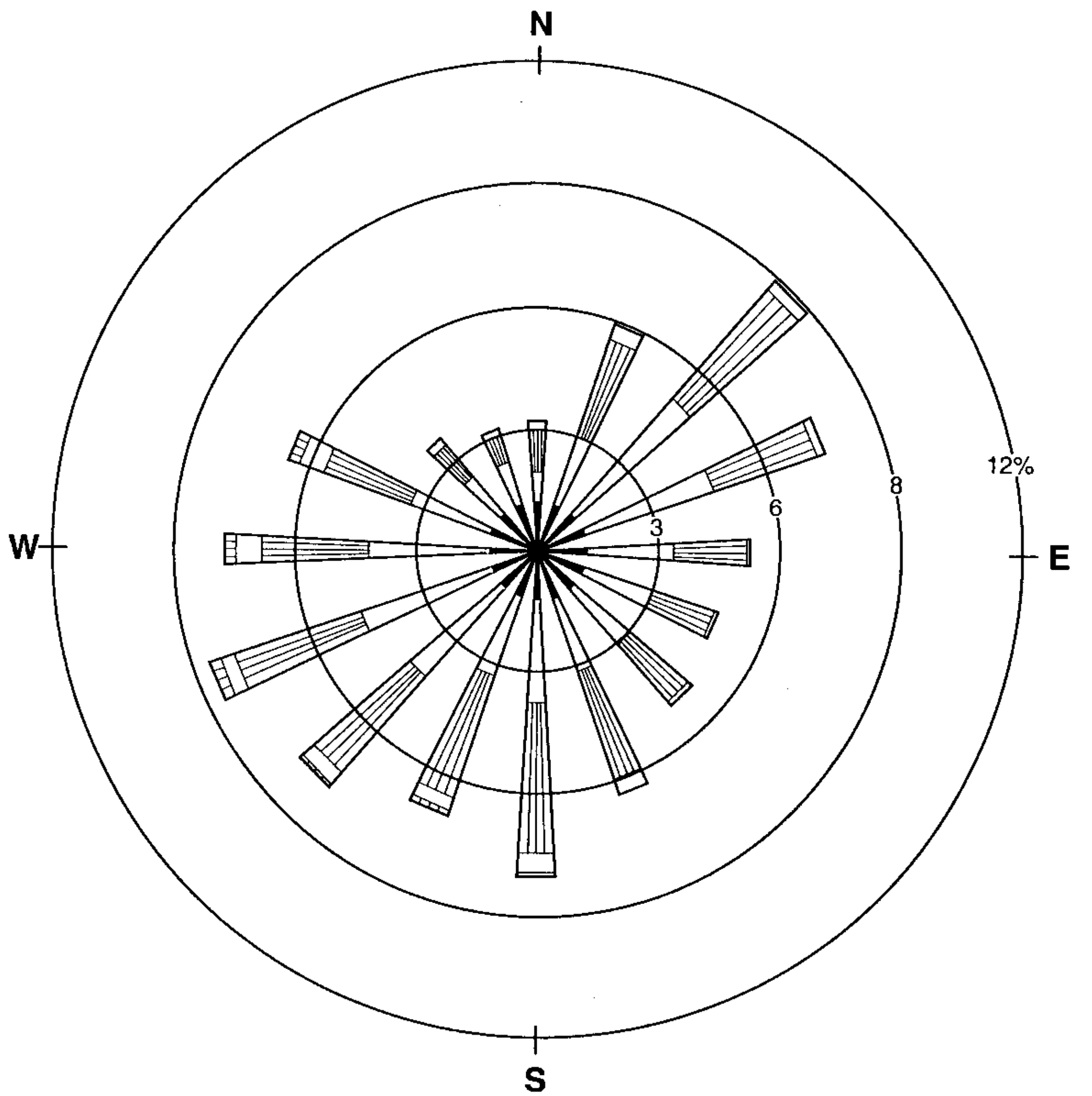


Figure 3-45 Groundwater Contamination at the Savannah River Site



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.

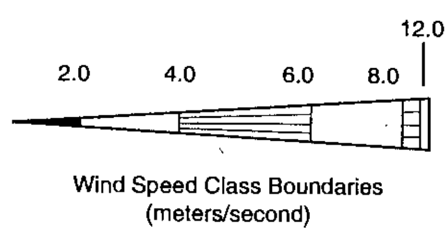


Figure 3-46 Wind Rose for the Savannah River Site (1987-1991)

Temperature and Humidity: The annual average temperature at the Savannah River Site is 17.8°C (64°F), and monthly averages range from a low of 7.22°C (45°F) in January to a high of 27.2°C (81°F) in July. Average daily relative humidity ranges from a maximum of 90 percent in the morning to a minimum of 43 percent in the afternoon on an annual basis.

Precipitation: The average annual precipitation at the Savannah River Site is approximately 121.9 cm (48 in). Precipitation distribution is fairly even throughout the year, with the highest precipitation in the summer [36.1 cm (14.2 in)] and the lowest in autumn [22.5 cm (8.8 in)] (Arnett et al., 1993). Snowfall has occurred in the months of October through March, with the average annual snowfall at 3.0 cm (1.2 in). Large snowfalls are rare (DOE, 1995c).

The area encompassing the Savannah River Site experiences an average of 56 thunderstorm days per year. From 1954 to 1983, 37 tornadoes were reported for a one-degree square of latitude and longitude that includes the Savannah River Site. 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 Savannah River Site is 0.00007 per year, which is less than one in ten thousand (DOE, 1995c). Since operations began at the Savannah River Site in 1953, nine tornadoes have been confirmed on or near the site. Winds exceeding hurricane force have been observed only once at the Savannah River Site (Hurricane Gracie in 1959) (Shedrow, 1993).

Atmospheric Dispersion: Based on measurements at onsite meteorological stations, dispersion conditions in the Savannah River Site region were classified 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 Savannah River Site (Shedrow, 1993).

Air Quality: The local air quality management region which includes the Savannah River Site is in attainment with National Ambient Air Quality Standards for criteria pollutants, which include sulfur dioxide, nitrogen oxides, particulate matter, lead, ozone (as volatile organic compounds), and carbon monoxide (EPA, 1993a). This region has a Class II designation under Prevention of Significant Deterioration regulations (EPA, 1993b), which allows moderate industrial growth to occur. No areas within an 80 km (50 mi) radius of the site are designated as Prevention of Significant Deterioration Class I (e.g., national parks, wildlife refuge). Class I areas place severe restrictions on new sources that might affect ambient air quality. The States of South Carolina and Georgia perform ambient air monitoring near the Savannah River Site, and have reported no significant exceedances of National Ambient Air Quality Standards.

In the Savannah River Site region, airborne radionuclides originate from natural resources (terrestrial or cosmic), worldwide fallout, and the Savannah River Site operations. The Savannah River Site 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 4-6 of Appendix C, Volume 1 of the Programmatic SNF&INEL Final EIS lists average and maximum atmospheric radionuclide particulate concentrations at the Savannah River Site boundary, and background [160 km (100 mi)] monitoring locations in 1991 (DOE, 1995c). Tritium is the only radionuclide of the Savannah River Site origin that can be detected routinely in offsite air samples above background.

3.3.1.5 Ecology

When the U.S. Government acquired the Savannah River Site in 1951, the site was approximately two-thirds forested and one-third cropland and pasture (Dukes, 1984). At present, more than 90 percent of the Savannah River Site is forested. With the exception of the Savannah River Site production and support areas, natural succession has reclaimed other previously disturbed areas. Satellite imagery of the site shows a circle of wooded habitat within a matrix of cleared uplands and narrow forested riparian corridors. The Savannah River Site provides nearly 73,250 ha (181,000 acres) of contiguous forested cover broken only by unpaved secondary roads, transmission line corridors, and a few paved primary roads. Carolina bays, the Savannah River swamp, and several relatively intact longleaf pine-wiregrass communities make important contributions to the biodiversity of the region.

The Savannah River Site 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). A variety of vascular plant communities occur in the upland areas. Typically, scrub oak communities occur on the drier, sandier areas. Longleaf pine, turkey oak, bluejack oak, blackjack oak, and dwarf post oak dominate these communities, which typically have understories of wire grass and huckleberry. Oak-hickory hardwood communities occur on more fertile, dry uplands, and characteristic species are white oak, post oak, southern red oak, mockernut hickory, pignut hickory, and loblolly pine, with an understory of sparkleberry, holly, greenbriar, and poison ivy (DOE, 1995c).

The Savannah River Site has provided excellent habitat to wildlife associated with the wetlands of the Savannah River and the pine-dominated sandhills of coastal South Carolina. Furbearers such as gray fox, raccoon, opossum and beaver are relatively common throughout the Savannah River Site. Game species such as gray and fox squirrel, cottontail rabbit, and wild turkey are also common. The Savannah River 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.

The Savannah River Site has extensive, widely distributed wetlands, most of which are associated with floodplains, creeks, and impoundments. The southwestern Savannah River Site boundary adjoins the Savannah River for approximately 32 km (20 mi). The river floodplain supports an extensive swamp, covering about 4,916 ha (12,148 acres) of the site. At present, the swamp forest consists of second-growth bald cypress, black gum, and other hardwood species (USDA, 1991). Five major streams drain the Savannah River Site, 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 the red maple, box elder, bald cypress, water tupelo, sweetgum, and black willow (DOE, 1995c). Carolina bays, unique wetland features of the southeastern United States, are islands of wetland habitat dispersed throughout the uplands of the Savannah River Site. The more than 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).

Threatened, Endangered, and Candidate Plant and Animal Species: Threatened, endangered, and candidate plant and animal species on the Savannah River Site include 5 bird species, 1 mammal species, 5 amphibian species, 5 reptile species, 1 fish species, 2 invertebrate species, and 19 plant species. The following Federally listed endangered animals are known to occur on the Savannah River Site or in the Savannah River adjacent to the site: the red-cockaded woodpecker, the southern bald eagle, the wood stork, and the shortnose sturgeon (DOE, 1995c). Researchers have found one Federally listed endangered plant species, the smooth coneflower, on the Savannah River Site, along with several Federally listed Category 2 species, and several listed species (Knox and Sharitz, 1990).

3.3.1.6 Land Use

The Savannah River Site occupies an area of approximately 800 km^2 (310 mi^2) in western South Carolina, in a generally rural area about 40 km (25 mi) southeast of Augusta, GA. The Savannah River Site, which is bordered by the Savannah River to the southwest, includes portions of Aiken, Barnwell, and Allendale

Counties. Land use on the Savannah River Site can be grouped into three major categories: forest/undeveloped, water/wetlands, and developed facilities. Ninety-six percent of the Savannah River Site area, about 73,450 ha (181,500 acres), is undeveloped (USDA, 1991). Approximately 90 percent of this area is forested (Cummins et al., 1990). In 1972, DOE designated the Savannah River Site as a National Environmental Research Park. At present, approximately 57 km^2 (22 mi^2), or 7 percent of the Savannah River Site area is designated as "Set-Asides," which are areas specifically protected for environmental research activities that are coordinated either through the University of Georgia Savannah River Ecological Laboratory or the Savannah River Technology Center (Cummins et al., 1990). At present, administrative production and support facilities occupy approximately 5 percent of the total the Savannah River Site land area.

Land bordering the Savannah River Site is primarily forest and agricultural. There is also a significant amount of open water and forested wetlands along the Savannah River Valley. Urbanized and industrial areas are the only other significant use of land in the vicinity (Figure 3-47). None of the three counties in which the Savannah River Site 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 the Savannah River Site, urban development and residential development. The closest residences to the Savannah River Site boundary include several within 61 m (200 ft) of the site perimeter to the west, north, and northeast.

The Savannah River Site is a controlled area, with public access limited to through traffic on South Carolina Highway 125 (the Savannah River Site Road A), U.S. Highway 278, the Savannah River Site Road 1, and the CSX railway. The Savannah River Site does not contain any public recreation facilities. However, the Savannah River Site conducts controlled deer and feral hog hunts each fall, from mid-October through mid-December. The intent of the hunts is to control the resident populations of these animals and to reduce animal-vehicle accidents on the Savannah River Site roads.

3.3.1.7 Noise

The major noise sources at the Savannah River Site are found 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 analyzed noise impacts of existing the Savannah River Site operational activities (DOE, 1995c; DOE, 1991a; DOE, 1990a; DOE, 1993d). These studies concluded that, because of the remote locations of the Savannah River Site 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 Savannah River Site as a result of hunting activities and construction activities. Noise limits are established for the workplace to protect workers' hearing in accordance with Occupational Health and Safety Administration standards. Existing Savannah River Site-related noise sources of importance to the public are those associated with the transportation of people and materials to and from the site. These sources include trucks, private vehicles, and freight trains. In addition, a portion of the air cargo and business travel using commercial air transport through the airports at Augusta, GA, and Columbia, SC, is attributable to the Savannah River Site operations. The States of Georgia and South Carolina, and the counties in which the Savannah River Site is located, have not established regulations that specify acceptable community noise levels, with the exception of a provision of the Aiken County Nuisance Ordinance, which limits daytime and nighttime noise by frequency band (Aiken County, 1991).

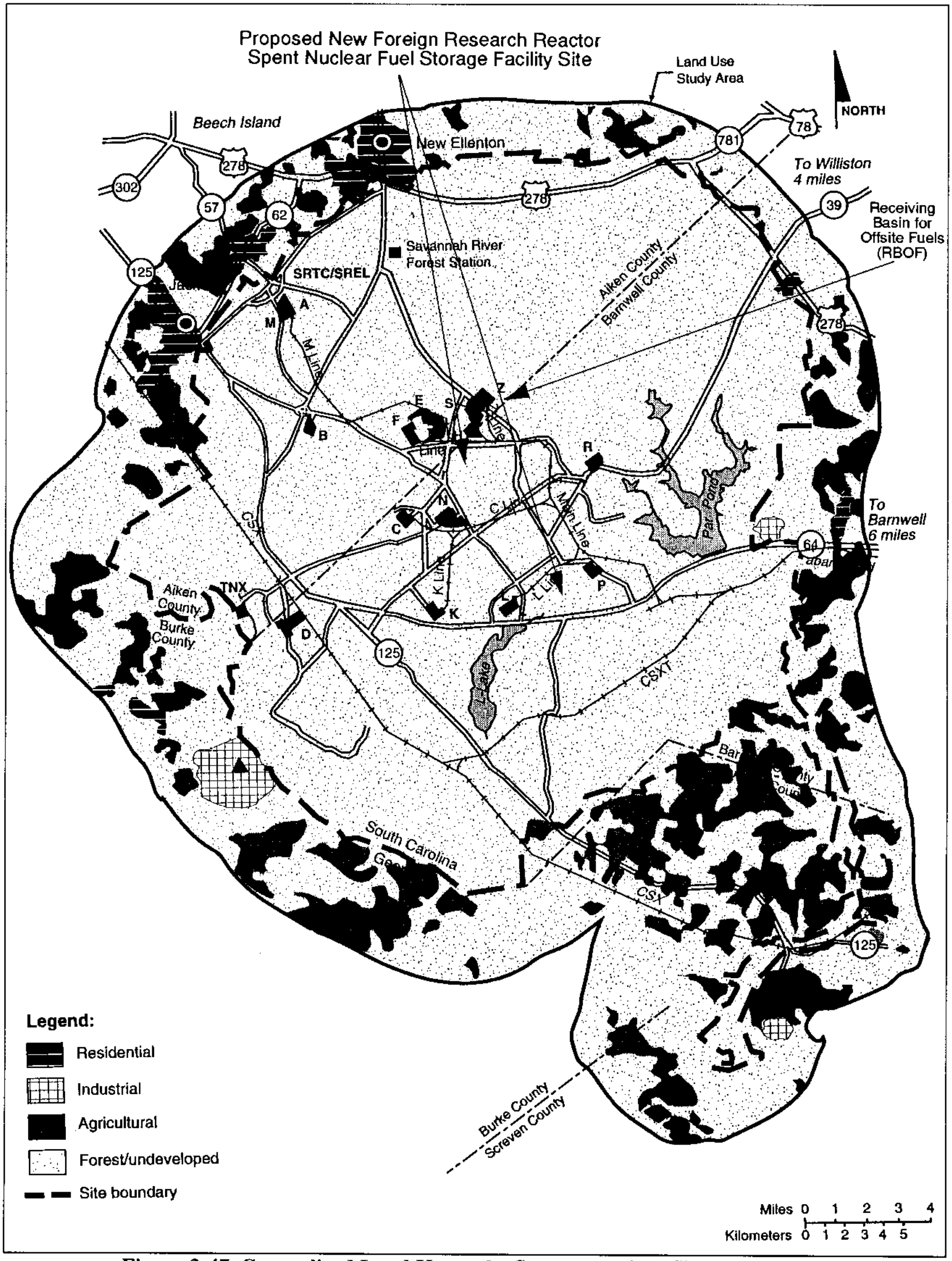


Figure 3-47 Generalized Land Use at the Savannah River Site and Vicinity

Noise from the Savannah River Site Traffic: During a normal week, about 20,000 employees travel to the Savannah River Site each day in private vehicles from surrounding communities. Both Government-owned and private trucks pick up and deliver materials at the site. The contribution of the Savannah River Site operations to traffic volumes along SC 125 and SC 19, especially during peak traffic periods, affects noise levels in 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 m or 50 ft from the roadway) indicate that the one-hour equivalent sound level from traffic ranged from 48 to 72 decibels. 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 indicate that the one-hour equivalent sound level from traffic ranged from 53 to 71 decibels. The estimated day/night average sound level along this route was 66 decibels for both summer and winter (HNUS, 1990). Employment at the Savannah River Site has increased by about 17 percent since 1989, potentially causing 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 major shift changes). Since some residences and at least two schools are within 30 to 60 m (100 to 200 ft) of these routes, some annoyance to members of the public residing along these highways can occur based on the relationship between the day/night average sound level (Schultz 1978; FICON, 1992).

Noise from Railroad Traffic: Approximately 18 trains per day pass through the Savannah River Site on the CSX line, with five trains delivering shipments to the Savannah River Site. Noise sources from rail transport include diesel engines, wheel-track contact, and whistle-warnings at rail crossings.

3.3.1.8 Transportation

The Savannah River Site 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 the Savannah River Site commuter traffic (DOE, 1995c). Two major railroads—CSX Transportation and Norfolk Southern Corporation—also serve the Savannah River Site vicinity. Norfolk Southern serves Augusta and Savannah, GA, as well as Columbia and Charleston, SC. CSX serves the same locations and the Savannah River Site. Figure 3-48 shows the regional transportation infrastructure.

Two Interstate highways serve the Savannah River Site area. Interstate 20 (I-20) provides a primary east-west corridor and 1-520 links I-20 with Augusta, GA. 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 Savannah River Site. From the northwest and north, access is provided by SC 125 and SC 19, respectively, and SC 125 is open to through traffic. Access to the site is provided from the northeast by SC 39, from the east by SC 64, and from the southeast by SC 125. These are all two-lane roads. The public has access to U.S. 278 and SC 125, but only the Savannah River Site employees are permitted access to the site on the other routes.

The Savannah River Site transportation infrastructure consists of more than 230 km (143 mi) of primary roads, 1,931 km (1,200 mi) of unpaved secondary roads, and 103 km (64 mi) of railroad track (DOE, 1995c). These roads and railroads provide connections among the various Savannah River Site facilities and offsite transportation linkages. Figure 3-49 shows the Savannah River Site network of primary roadways and access points, and the Savannah River Site railway system.

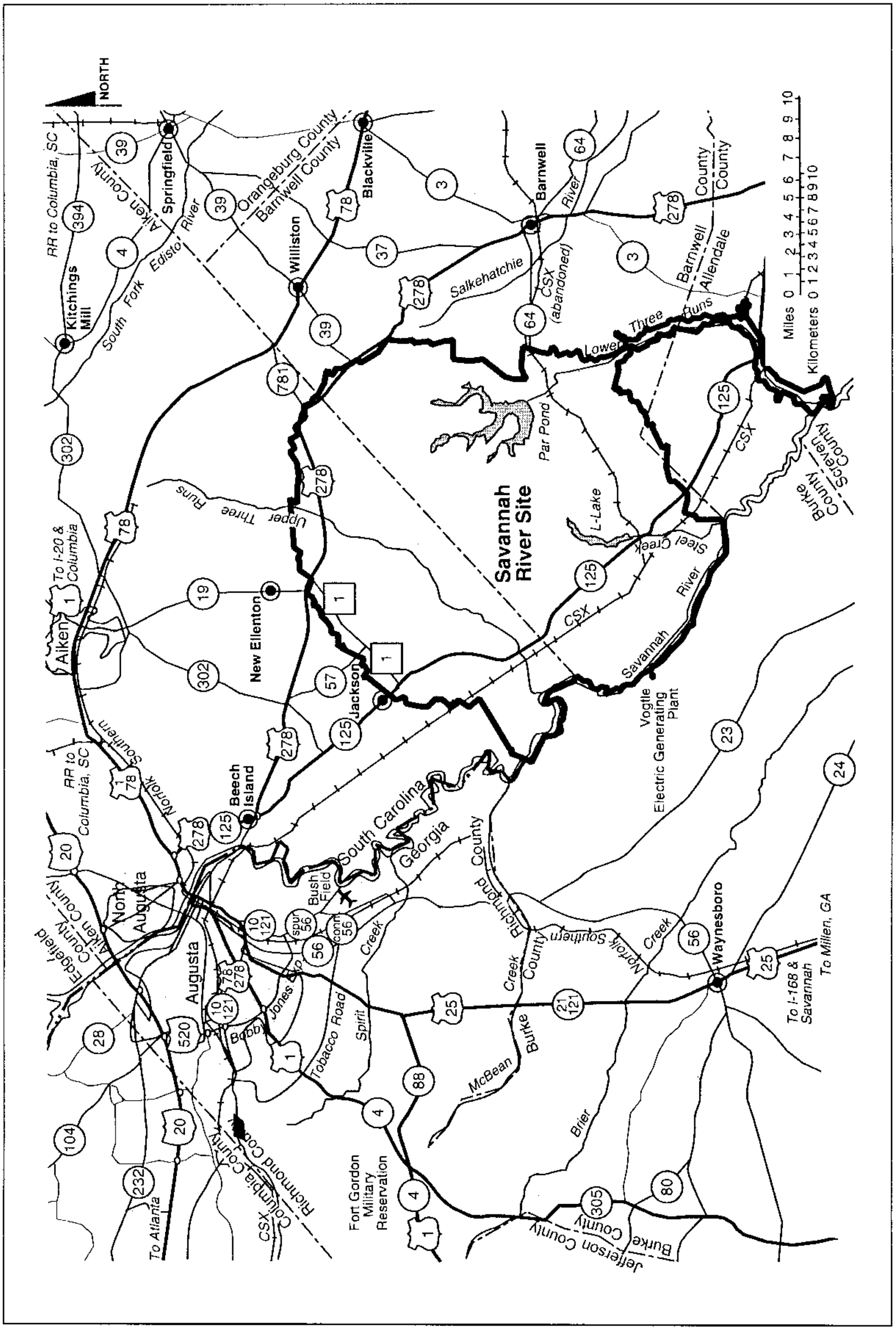


Figure 3-48 Regional Transportation Infrastructure

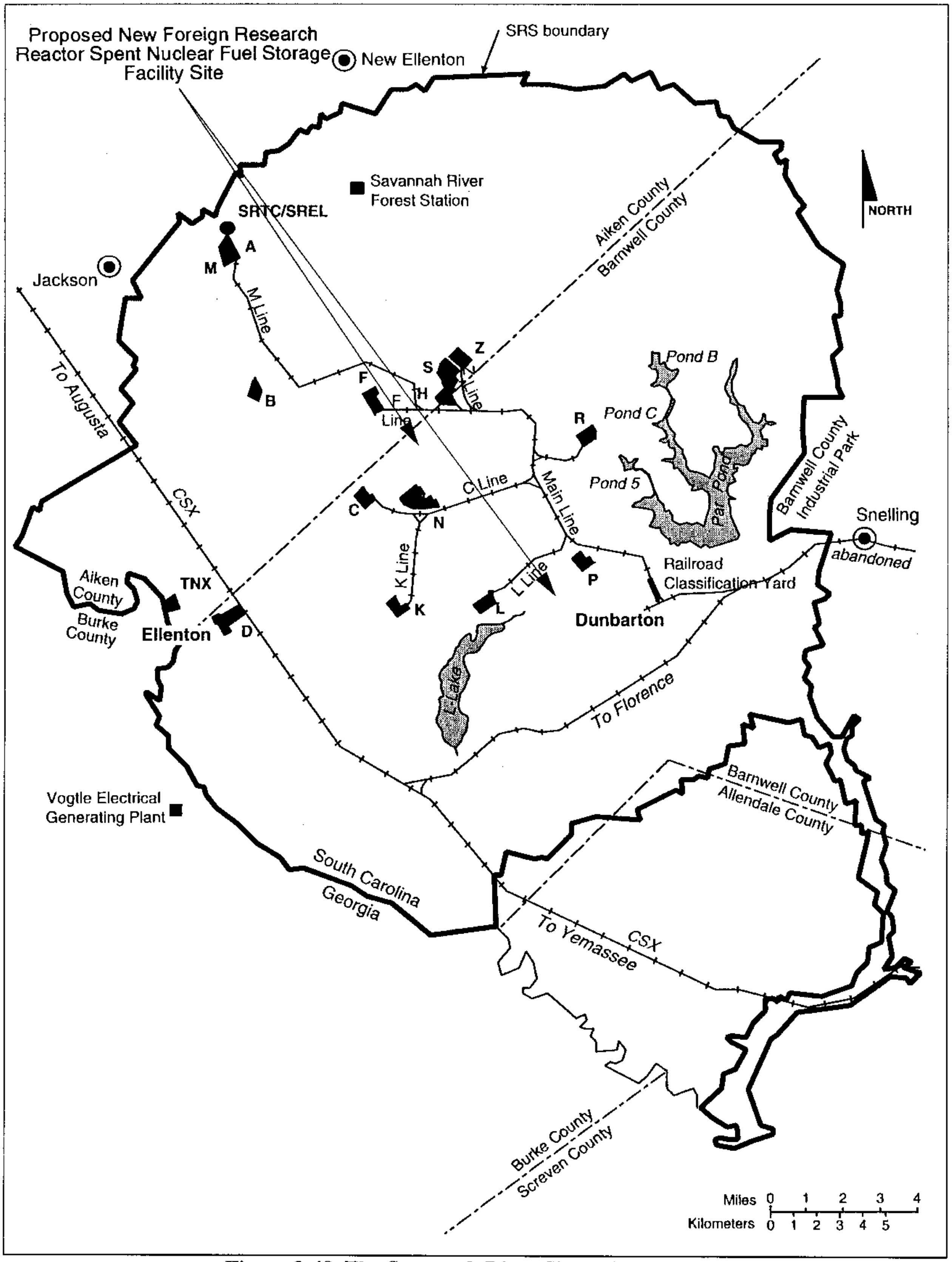


Figure 3-49 The Savannah River Site Railroad Lines

Two major public highways traverse the Savannah River Site: SC 125 and U.S. 278. SC 125 connects Allendale, SC, to Augusta, GA, by crossing the site in a northwest-to-southeast direction. U.S. 278 also connects Augusta and Allendale, but its route generally follows the northern and eastern Savannah River Site boundaries. In general, the primary Savannah River Site 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.

In general, heavy traffic occurs early in the morning and late in the afternoon when workers from surrounding communities commute to and from the Savannah River 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 Savannah River Site, with an annual average of about 25 trucks per day. Table 4-16 of Appendix C, Volume 1 of the Programmatic SNF&INEL Final EIS provides data on traffic counts for various roads and access points around the Savannah River Site (DOE, 1995c).

Railroads on the Savannah River Site include both CSX tracks and the Savannah River Site rolling stock and tracks. Two routes of the CSX distribution system run through the Savannah River Site: a line between Florence, SC, and Augusta, GA, and a line between Yemassee, SC, and Augusta. The two lines join on the site near the L-Lake dam (Figure 3-49). Early in 1989, CSX discontinued service on the line from the Savannah River Site junction to Florence. The 103 km (64 mi) of the Savannah River Site railroad tracks are well maintained. The rails and crosslines 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. The Savannah River Site rail classification yard is east of P-Reactor. This eight-track facility sorts and redirects railcars. Deliveries of the Savannah River Site shipments occur at two onsite rail stations at the former towns of Ellenton and Dunbarton. From these stations, a Savannah River Site 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.

3.3.1.9 Socioeconomics

The Savannah River Site region of influence includes Aiken, Allendale, Bamberg, and Barnwell Counties in South Carolina, and Columbia and Richmond Counties in Georgia. 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 of Appendix C, Volume 1 of the Programmatic SNF&INEL Final EIS lists projected employment data for the six-county region of influence. As shown, by the year 2000, employment levels should increase 27 percent to approximately 253,000. The unemployment rates for 1980 and 1990 were 7.3 percent and 4.7 percent, respectively (DOE, 1995c).

In 1990, employment at the Savannah River Site was 20,230, representing 10 percent of the region of influence employment (DOE, 1993d). In Fiscal Year 1992, employment at the Savannah River Site increased approximately 15 percent to 23,351, with an associated payroll of more than $1.1 billion. From 1980 to 1990, the labor force in the six-county region of influence grew 39 percent, from 150,551 to 208,984. In 1990, 75.3 percent of the region of influence labor force lived in Richmond and Aiken Counties, SC. Current projections call for the region's labor force to increase to approximately 257,000 workers by 1995 (DOE, 1995c).

Between 1980 and 1990, 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. According to 1990 census data, 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 (DOE, 1995c). Based on 1990 census population data, the general ethnic composition of the immediate area of influence, which is within an 80 km (50 mi) radius of the site, is shown in Figure 3-50. Low-income households are presented in Figure 3-51. Low-income households are those with incomes of 80 percent or less than the median income of the counties. As indicated in this figure, approximately 42 percent of the total households are low-income households.

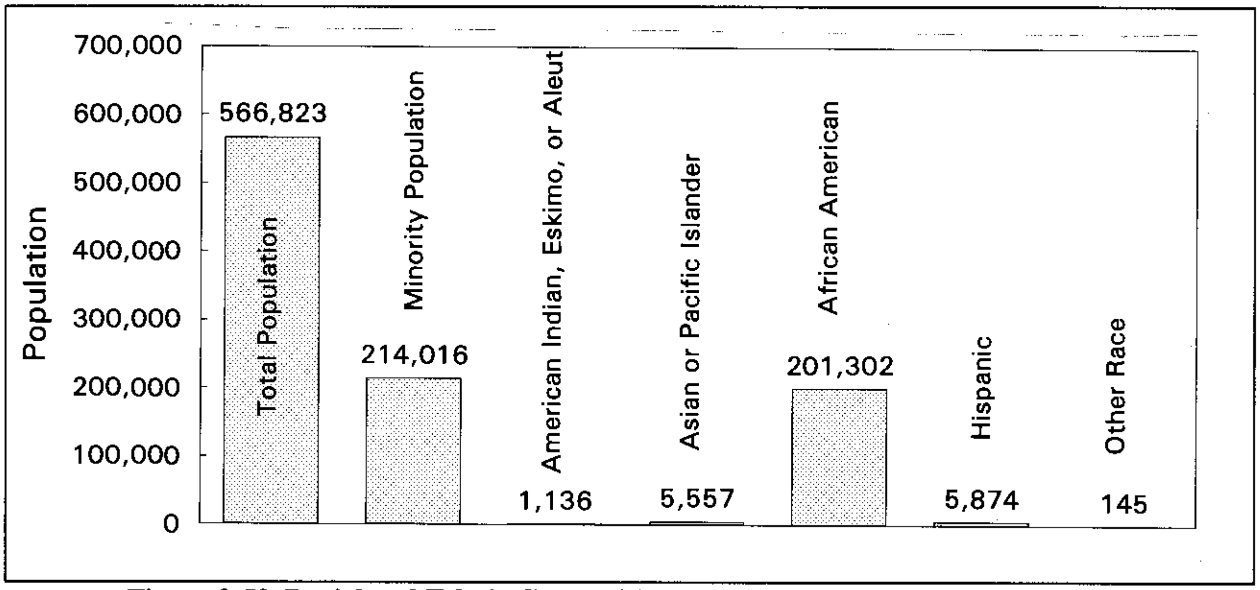


Figure 3-50 Racial and Ethnic Composition of the Minority Population Residing within 80 km (50 mi) of the Savannah River Site

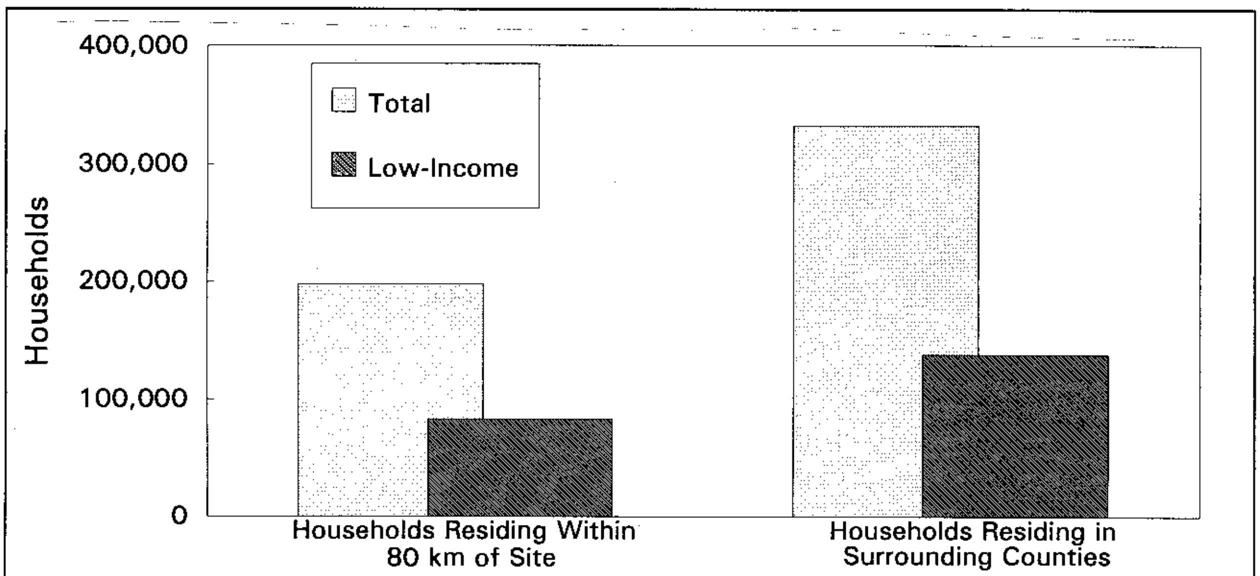


Figure 3-51 Low-Income Households Residing within 80 km (50 mi) of the Savannah River Site

3.3.1.10 Historical, Archaeological and Cultural Resources

By the end of Fiscal Year 1992, approximately 60 percent of the 800 km^2 (310 m^2) of the Savannah River Site had been examined, and 858 archaeological sites had been identified. Of these, 53 have been determined to be eligible for the National Register of Historic Places, and 650 have not been evaluated (DOE, 1995c).

Three Native American groups, the Yuchi Tribal Organization, the National Council of Muskogee Creek, and the Indian People's Muskogee Tribal Town Confederacy, have expressed concerns over sites and items of religious significance on the Savannah River Site.

Archaeologists have divided areas of the Savannah River Site into three sensitivity zones related to their potential for containing sites with multiple archaeological components or dense or diverse artifacts, and their potential for eligibility to the National Register of Historic Places (DOE, 1995c).

Zone One is the zone of highest archaeological site density, with a high probability of encountering large archaeological sites with dense and diverse artifacts, and high potential for nomination to the National Register of Historic Places. Zone Two covers areas of moderate archaeological site density that should contain sites of similar composition. Activities in this zone have a moderate probability of encountering archaeological 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 Three covers areas of low archaeological site density. Activities in this zone have a low probability of encountering archaeological sites and virtually no chance of encountering large sites with more than three prehistoric components. Therefore, 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.

3.3.2 Description of the Affected Environment at the Idaho National Engineering Laboratory

This section describes the potentially affected environment of the Idaho National Engineering Laboratory. The location of the site is shown in Figure 3-52.

3.3.2.1 Geology

The Idaho National Engineering Laboratory is located within a broad low-relief basin floored with basaltic lava flows and terrestrial sediments in the Eastern Snake River Plain physiographic province. The Snake River Plain extends in a broad arc from the Idaho-Oregon border in the west to the Yellowstone Plateau in the east, and contrasts sharply with the surrounding mountainous country of the Northern Basin and Range Province and the Idaho Batholith.

The Snake River Plain was formed in response to the movement of the North American Continent over a deep seated plume of hot mantle rocks. Movement of the continent and a northeast directed extension of the crust caused development of both the Eastern Snake River Plain and the northern Basin and Range Province over the past 17 million years. This movement has produced northwest trending normal faults in the Basin and Range Province and volcanic activity, including the formation of calderas, rhyolite domes, and volcanic rifts or vents.

Surface rocks on and near the Idaho National Engineering Laboratory are mostly an interlayered sequence of basalt flows and poorly consolidated sedimentary interbeds to a depth of 1 to 2 km (0.6 to 1.2 mi). Interbedded sediments are composed mostly of fine-grained silts and clays. A wide band of Quaternary

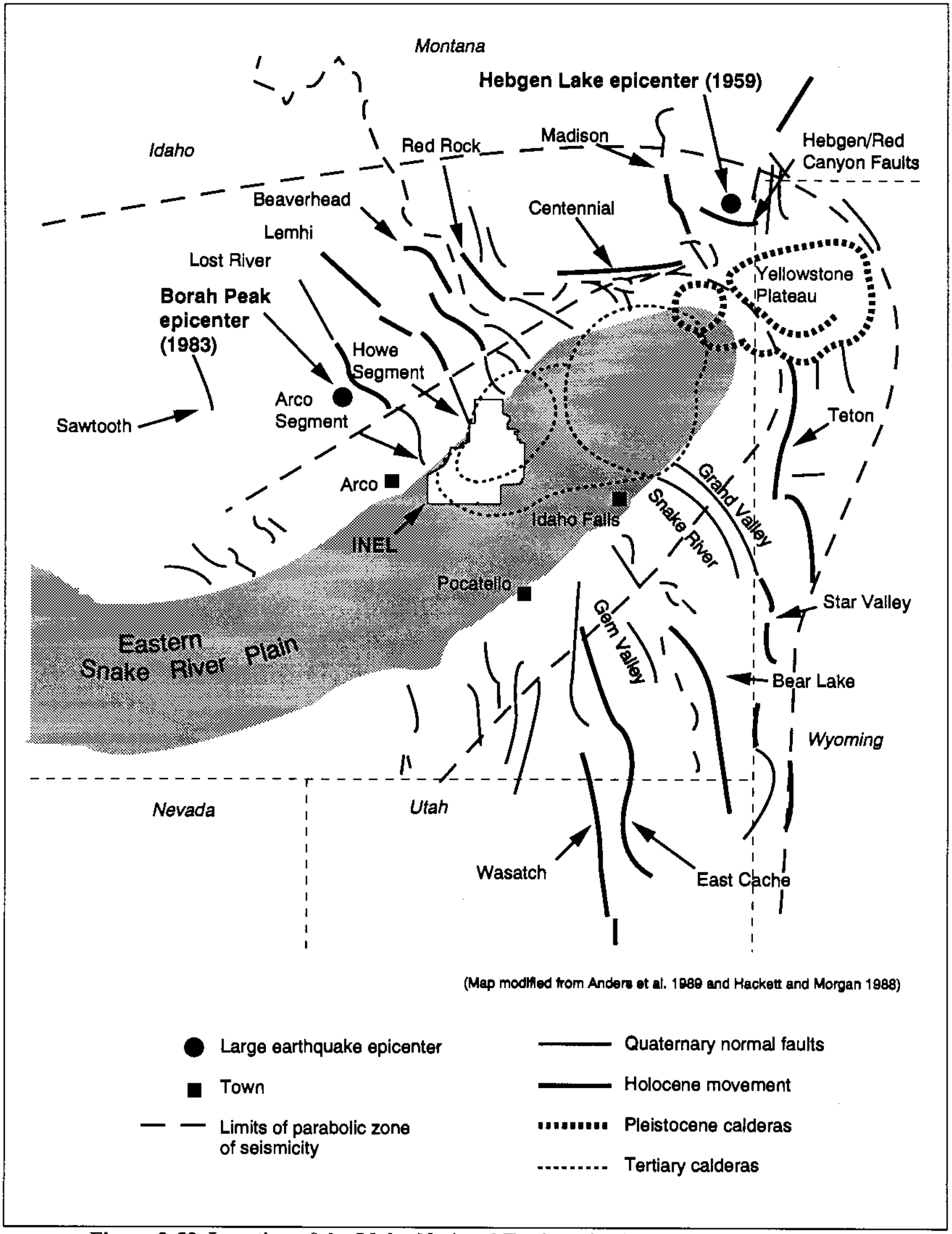


Figure 3-52 Location of the Idaho National Engineering Laboratory in Context of Regional Geologic Features

mainstream alluvium extends along the course of the Big Lost River (southwestern corner of the Idaho National Engineering Laboratory) to the Lost River Sinks area in the north-central portion of the Idaho National Engineering Laboratory. Lacustrine clay and sand deposits from the Ice Age Lake Terreton occur in the northern part of the site. Some of these beach sands were reworked by winds in late Pleistocene and Holocene times to form large dune fields in the northeastern part of the Idaho National Engineering Laboratory (Scott, 1982). Several Quaternary rhyolite domes occur along the Axial Volcanic Zone near the south and southeast borders of the Idaho National Engineering Laboratory. Paleozoic limestones, late-Tertiary rhyolitic volcanic rocks, and large alluvial fans occur in limited areas along the northwest margin of the site.

3.3.2.2 Seismology and Volcanology

The effusion of basaltic-lava flows from volcanic rift zones has been the predominant style of volcanism on the Eastern Snake River Plain, including the Idaho National Engineering Laboratory area, during the past 4 million years. Broad uplift of the ground surface, together with the opening of fissures and faults, accompany the ascending magma before eruptions. Vents for the basaltic volcanism are concentrated in northwest-trending volcanic rift zones and along the Axial Volcanic Zone (Kuntz, 1992).

Volcanic vents on the Eastern Snake River Plain are concentrated in several linear belts that trend northwest and northeast. These northwest-trending belts are associated with ground deformation features referred to as volcanic rift zones with fissures, faults, grabens, and monoclines. Volcanic vents are also concentrated in a northeast-trending zone along the axis of the Eastern Snake River Plain, called the Axial Volcanic Zone to distinguish it from volcanic rift zones (Hackett and Smith, 1992). Basaltic-lava eruptions appear to have been most frequent and most recent (approximately 5,200 years ago at Hell's Half Acre) along this zone, and at intersections of that zone with northwest-trending volcanic rift zones in the site area. Volcanic vents in this area are fed by northwest-trending dikes with seismicity associated with volcanism.

The Eastern Snake River Plain is surrounded by the seismically active Intermountain Seismic and Centennial Tectonic Belts (Smith and Arabasz, 1991). The Snake River Plain is devoid of earthquakes relative to the active areas surrounding it. The Eastern Snake River Plain has exhibited infrequently-occurring small magnitude (M 1.5) earthquakes.

3.3.2.3 Hydrology

3.3.2.3.1 Surface Water

Surface water at the Idaho National Engineering Laboratory accumulates into streams from local rainfall, snowmelt, and runoff originating in the mountain ranges located directly north and west of the Idaho National Engineering Laboratory. Surface water also includes water contained in man-made infiltration and evaporation ponds. Except for standing water in man-made ponds, there is little surface water at the Idaho National Engineering Laboratory. However, during wet years, when runoff from drainage basins or snowmelt is heavy, surface water bodies are formed.

The Idaho National Engineering Laboratory is located in Pioneer Basin, a closed drainage basin that includes three main surface water bodies: The Big Lost River, the Little Lost River, and Birch Creek. These sources of water drain mountain watersheds located north and west of the Idaho National Engineering Laboratory. However, most of the surface water flow is diverted for irrigation before it reaches site boundaries (Barraclough et al., 1981). This has resulted in little or no flow in these surface water bodies for several years within the boundaries of the Idaho National Engineering Laboratory (Pittman et al., 1988).

The Big Lost River is the major surface water body at the Idaho National Engineering Laboratory. The river flows between the Lost River Range and the Pioneer Mountains, draining approximately 3,755 km^2 (1,450 mi^2) of land before reaching the site. Approximately 48 km (30 mi) upstream of Arco, ID, Mackay Dam controls and regulates the flow of the river. During heavy runoff events, surface water from the Big Lost River is diverted to four spreading areas located south of the Idaho National Engineering Laboratory Diversion Dam. North of the diversion dam, the Big Lost River continues northward across the site to an area of natural infiltration basins (playas or sinks) near Test Area North. Flow within the Big Lost River channel may only reach a few kilometers southeast of Arco during drought years.

Birch Creek flows through an elongated, southeast-trending valley located between the Lemhi and Beaverhead Mountain Ranges, which cross the northwest corner of the Idaho National Engineering Laboratory to Playa 4 near Test Area North. Birch Creek drains an area of approximately 1,943 km^2 (750 mi^2). In the summer, upstream of the Idaho National Engineering Laboratory, surface water from Birch Creek is diverted for irrigation and to produce hydropower. In the winter, water flows in a man-made channel constructed 6.4 km (4 mi) north of Test Area North, where it infiltrates into channel gravel.

The Little Lost River drains the slopes of the Lemhi and Lost River Ranges, an area of approximately 1,826 km^2 (705 mi^2). Surface water from the Little Lost River has not reached the Idaho National Engineering Laboratory in recent times; during high streamflow years, however, water will reach the site and infiltrate into the subsurface (EG&G Idaho Inc., 1984).

Surface water generated from local precipitation will flow into topographic depressions (lower elevations than the surrounding terrain). Surface water within the depressions either evaporates or infiltrates into the ground. Localized flooding can occur at the Idaho National Engineering Laboratory when the ground is frozen and melting snow is combined with heavy spring rains. Other facility areas have been threatened by flooding because of ice jams in the Big Lost River and its diversion channel (McKinney, 1985).

Intermittent surface flow and the Idaho National Engineering Laboratory Diversion Dam have effectively prevented the Big Lost River from flooding onto the site. Flood plains in existence before the diversion dam and channels were built are not active. Based on historical data from past storm events, if heavy runoff occurs and surface water flow from the Big Lost River is diverted to the four spreading areas, water will not overflow the banks of the existing Big Lost River channel onsite during 100- and 500-yr floods (floods that occur on an average of every 100 or 500 years).

Onsite flooding from the Big Lost River may occur if high water in the MacKay Dam or the Big Lost River is coupled with a dam failure. Assuming the Mackay Dam fails, significantly higher flows and greater flooding would be associated with the probable maximum flood than either the 100- or 500-year floods (Figure 3-53).

Surface Water Quality: Water quality in the Big Lost River, Little Lost River, and Birch Creek is similar, and has not varied significantly over the period of record. The chemical composition of these water bodies is determined primarily by the mineral composition of the rocks in surrounding mountain ranges northwest of the site, and by the chemical composition of irrigation water in contact with the surface water

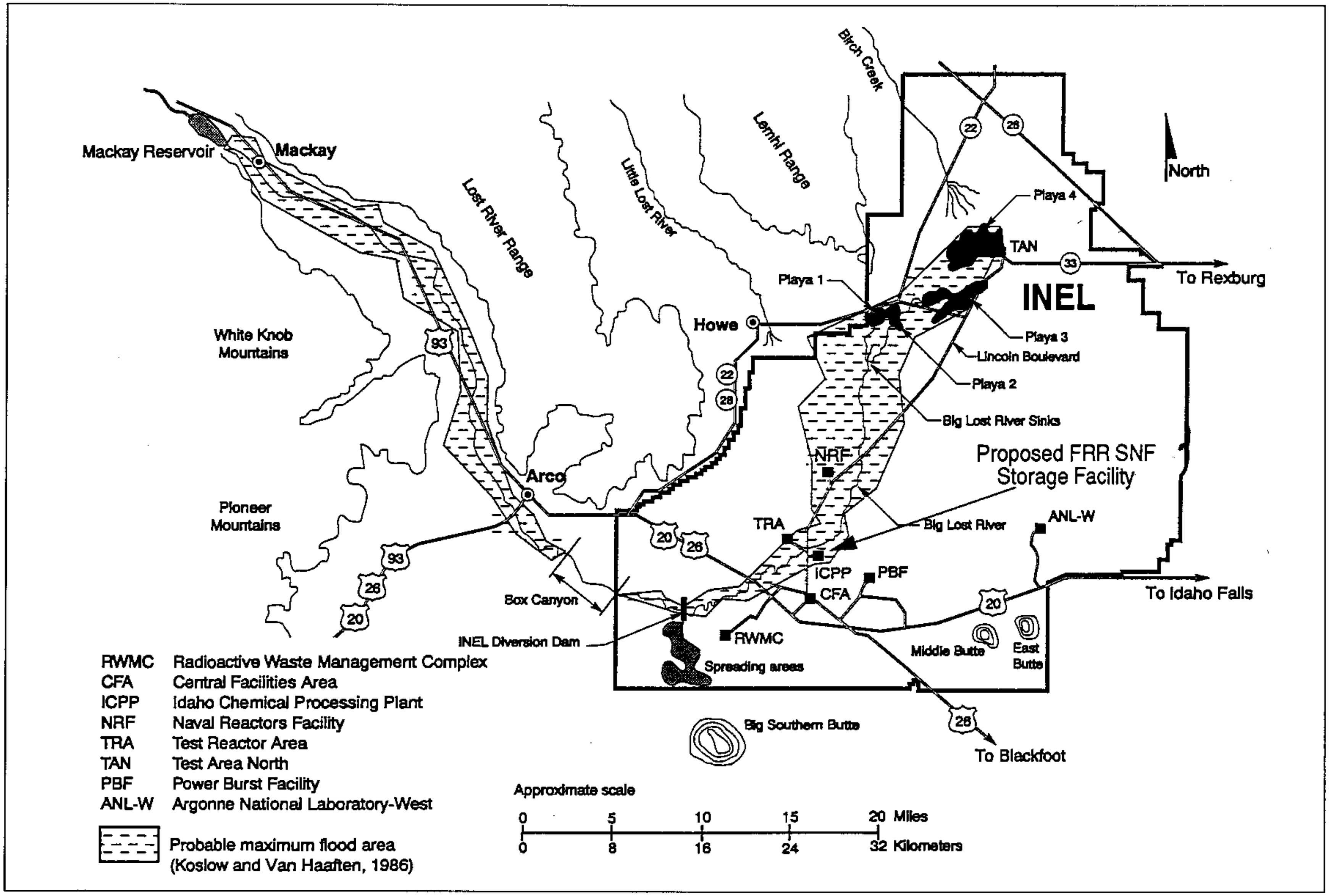


Figure 3-53 Selected Facilities and Predicted Inundation Map for Probable Maximum Flood-Induced Overtopping Failure of Mackay Dam at the Idaho National Engineering Laboratory

(Robertson et al., 1974; Bennett, 1990). Chemical and physical parameters measured in the Big Lost River, Little Lost River, and Birch Creek do not exceed drinking water quality standards for the parameter analyzed (DOE, 1995c).

The Idaho National Engineering Laboratory activities do not directly affect the quality of surface water outside the site, because discharges from the Idaho National Engineering Laboratory facilities are made to man-made seepage and evaporation basins or stormwater injection wells. Surface water is not withdrawn by humans for use at the Idaho National Engineering Laboratory, and discharge of effluents to natural surface waters does not occur. In addition, surface water does not flow directly offsite (Hoff et al., 1990). However, water from the Big Lost River, as well as from seepage of evaporation basins and stormwater injection wells, does infiltrate into the Snake River Plain Aquifer, and will eventually reach the Snake River (Robertson et al., 1974; Wood and Law, 1988; Bennett, 1990).

3.3.2.3.2 Groundwater

Groundwater at the site occurs in the Snake River Plain Aquifer and the vadose zone. The Idaho National Engineering Laboratory overlies the Snake River Plain Aquifer, which covers an area of approximately 24,900 km^2 (9,614 mi^2). Groundwater in the aquifer generally flows south and southwestward across the Snake River Plain, discharging into the Snake River at the Thousand Springs Area near Twin Falls, ID.

The Snake River Plain Aquifer is recharged by seepage of irrigation water, stream channel and canal leakage, tributary drainage basin underflow, and direct infiltration from precipitation (Garabedian, 1989). The drainage basin recharging the aquifer covers an area of approximately 90,643 km^2 (35,000 mi^2). Most recharge occurs in surface water-irrigated areas and along the northeastern margins of the plain. A majority of the groundwater discharged from the aquifer is through springs that flow into the Snake River, and through pumping for irrigation purposes. 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, ID).

Water storage in the Snake River Plain Aquifer is estimated at 2.5 x $10^{+12}m^3$ (8.8 x $10^{+13}ft^3$) (Robertson et al., 1974). Irrigation wells yield as much as 26.5 m^3 per min (7,000 gal per min) of water (Garabedian, 1989). Groundwater discharges primarily from the aquifer through springs that flow into the Snake River and from pumping for irrigation purposes. Major springs and seepages that flow from the aquifer are localized near the American Falls Reservoir (southwest of Pocatello) and the Thousand Springs area between Milner Dam and King Hill.

The Idaho National Engineering Laboratory covers 2,305 km^2 (890 mi^2) of the north-central portion of the Snake River Plain Aquifer. Most of the aquifer is composed of a thick sequence of relatively thin basaltic lava flows with sedimentary interbeds extending to depths greater than 1,067 m (3,500 ft) below the land surface (Irving, 1993). The basalt flows are interbedded with sedimentary layers formed during periods between volcanic eruptions when the basalt was exposed and sediments collected on the land surface. A majority of the groundwater moves horizontally through fractured, basaltic interflow zones (broken and rubble zones) that occur at various depths. Water also moves vertically along joints and the interfingering edges of interflow zones (Garabedian, 1986). Sedimentary interbeds restrict the vertical movement of groundwater.

Depths to the water table from the land surface at the Idaho National Engineering Laboratory range from approximately 61 m (200 ft) in the north, to more than 274 m (900 ft) in the south (Pittman et al., 1988). The upper surface of the aquifer is unconfined over most of its extent. However, the aquifer generally behaves as if it were partially confined, because water moves through dense basalt with interbedded sediments and water-bearing basaltic fracture zones at different rates. The base of the aquifer coincides with the tip of a thick, widespread sequence of clay, silt, sand, and basalt that occurs at depths ranging from 244 to 457 m (800 to 1,500 ft) below the land surface. The thickness of the aquifer is primarily controlled by the geologic setting, and therefore varies across the Idaho National Engineering Laboratory (Anderson, 1991).

The rate water moves through the aquifer depends on the gradient (change in elevation with distance) of the water table, the porosity of the soil and bedrock (void spaces in aquifer materials), and the hydraulic conductivity of the soil and bedrock (capacity of a porous media to transport water). Across the Idaho National Engineering Laboratory, the horizontal gradient of the water table ranges from 0.19 to 2.9 m/km (1 to 15 ft/mi) (Ackerman, 1991). Vertical hydraulic gradients (change in elevation, pressure, and velocity with distance in a given direction) are usually less than 0.01 m/m (0.01 ft/ft) in the first 61 m (200 ft) below the land surface, and less than 0.02 m/m (0.02 ft/ft) in the first 168 m (550 ft) of the saturated thickness. Groundwater flows horizontally at velocities ranging from 1.5 to 7.6 m/day (5 to 25 ft/day). However, most of the water flows from 1.5 to 3 m/day (5 to 10 ft/day) (Robertson et al., 1974).

Transmissivity values vary widely across the Idaho National Engineering Laboratory. Data from 183 single-well tests at 94 wells provide estimates of transmissivity ranging from 0.1 to 70,604 m^2 per day (1.1 to 760,000 ft^2 per day) (Ackerman, 1991). The lowest transmissivity rates are generally at the northern end of the Idaho National Engineering Laboratory, and the highest rates are near the Test Reactor Area in the south-central part of the Idaho National Engineering Laboratory. Aquifer storativity is estimated using results from multiple-well aquifer tests. Values of storativity (or storage coefficients) range from 0.0I to 0.06, indicating generally unconfined aquifer conditions at the site. The aquifer behaves as an unconfined system, which increases well yields. However, clay layers and dense, unfractured basalts in the aquifer are locally confining.

Since aquifer porosity and hydraulic conductivity decrease with depth, most of the water in the aquifer moves through the upper 61 to 152 m (200 to 500 ft) of the Quaternary basalts. Estimated flow rates within the aquifer range from 1.5 to 6.1 m (5 to 20 ft) per day. Recharge to the aquifer near the Idaho National Engineering Laboratory originates from precipitation in the mountains to the north and west. Lesser amounts of recharge occur from local precipitation and snowmelt (Barraclough et al., 1981).

The vadose zone at the Idaho National Engineering Laboratory extends from the land surface down to the water table, with a thickness ranging from 61 m (200 ft) in the north, to greater than 274 m (900 ft) in the south. The vadose zone consists of surface sediments and relatively thin, basaltic lava flows with occasional interbedded sediments. The surface sediments are composed of clay, silt, sand, and gravel. Thick surficial deposits of (primarily) clay and silt are found in the northern portion of the Idaho National Engineering Laboratory, and thin deposits are found southward, where basalt is exposed at the surface.

Perched water at the site generally occurs under presence of disposal ponds or other surface water features. Perched water bodies have been detected at the Idaho Chemical Processing Plant, Test Reactor Area, Test Area North, and Radioactive Waste Management Complex facility areas.

Groundwater Quality: Previous waste discharges to unlined ponds and deep wells have introduced radionuclides, nonradioactive metals, inorganic salts, and organic compounds to the subsurface (DOE, 1995c).

Radionuclide concentrations in the Snake River Plain Aquifer beneath the Idaho National Engineering Laboratory have decreased since the mid-1980's because of changes in disposal practices, radioactive decay, adhesion 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). 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, 1993).

The Idaho National Engineering Laboratory has released sodium, chromium, lead, and mercury on the site and into the subsurface through unlined ponds and deep wells. Sodium is the greatest quantity of material released from the waste treatment processes. However, sodium is not toxic. In 1988, chromium concentrations exceeding the maximum allowable 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).

Concentrations of volatile organic compounds have been detected in the aquifer beneath the Idaho National Engineering Laboratory. 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; Liszewski and Mann, 1993).

Groundwater uses on the Snake River Plain include irrigation, food processing and agriculture, and domestic, rural, public, and livestock supply. Water use for the upper Snake River drainage basin and Snake River Plain Aquifer was 16.4 x 10^{+10} cubic m (4.3 x 10^{+12} gal) per year during 1985, which was more than 50 percent of the water used in Idaho, and approximately 7 percent of agricultural withdrawals in the nation. Site activities withdraw water at an average rate of 7,400,000 m^3 (1.9 x 10^{+9} gal) per year. This rate is equal to approximately 0.4 percent of the water consumed in the Eastern Snake River Plain Aquifer (DOE, 1995c).

Since 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 U.S. Environmental Protection Agency designated the Snake River Plain Aquifer a sole-source aquifer in 1991 (EPA, 1991b).

The Idaho Department of Water Resources manages groundwater resources to meet the State's water needs. Idaho operates under the system of appropriation rights — the senior appropriation has priority in times of shortage, or "first in time, first in right."

DOE holds a Federal Reserved Water Right for the Idaho National Engineering Laboratory, which permits a water pumping capacity of 2.3 m^3 per sec (80 ft^3 per sec), and a maximum water consumption of 43 million m^3 per year (11.4 billion gal per year) for drinking, process water, and noncontact cooling. Because it is a Federal Reserved Water Right, the Idaho National Engineering Laboratory's priority on water rights dates back to the establishment of the Idaho National Engineering Laboratory.

3.3.2.4 Meteorology

The Eastern Snake River Plain climate exhibits low relative humidity, wide daily temperature swings, and large variations in annual precipitation. Several topographic characteristics of the area (including altitude, latitude, and intermountain setting) influence the climate of the site area.

Wind: The terrain impacts the regional wind flow. The Idaho National Engineering Laboratory is in the belt of prevailing westerlies. Nighttime winds are common at the Idaho National Engineering Laboratory on clear or partly cloudy nights. These winds blow from the northeast and are formed by the rapid rotational cooling of the near-ground air layer on mountain slopes. Wind directions with the highest frequency of occurrence as measured onsite, are from south to west-southwest and from northwest to northeast (Clawson et al., 1989).

The highest hourly average near-ground (6.1 m, 20 ft level) windspeed measured onsite is 22.8 m per sec (51 mph) from the west-southwest. The maximum instantaneous gust at this level was 34.9 m per sec (78 mph). Some of the highest windspeeds occur in the spring, which coincides with strong prevailing westerlies at higher altitudes (Clawson et al., 1989). Apart from thunderstorms, severe weather is uncommon. Visibility in the region is good due to the low moisture content of the air and minimal sources of visibility-reducing pollutants. The background visibility is estimated at 60 km (37.5 mi).

Temperature and Humidity: Average seasonal temperatures measured onsite 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 measured at the Idaho National Engineering Laboratory range from a summertime maximum of 39.4°C (103°F), to a wintertime minimum of -45°C (-49°F). Temperature differences in excess of 13.9°C (25°F) have been observed onsite (Clawson et al., 1989).

The annual average relative humidity at the Idaho National Engineering Laboratory 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).

Precipitation: Annual precipitation is light, averaging 22.12 cm (8.71 in). The monthly average precipitation peaks in May and June. For the rest of the year, precipitation is uniformly distributed, averaging 1.27 to 1.78 cm (0.5 to 0.7 in) monthly. The highest monthly average snowfall of 16.26 cm (6.4 in) occurs in December, with similar amounts during January. The average annual snowfall is 70.1 cm (27.6 in). Maximum annual snowfall is 151.6 cm (59.7 in), and minimum annual snowfall is 17.3 cm (6.8 in) (Clawson et al., 1989).

Atmospheric Dispersion: Vertical diffusion of pollutants may be restricted or enhanced by the vertical temperature gradient of the atmosphere (that is, lapse or inversion conditions). These inversions are often ground-based, meaning that the temperature increases with height from the ground (Clawson et al., 1989).

Air Quality: The population of the Eastern Snake River Plain is exposed to environmental radiation from both natural and man-made sources.

Background radiation includes sources such as cosmic rays, radioactivity naturally present in soil, rocks, and the human body, and airborne radionuclides of natural origin (such as radon). Radioactivity still remaining in the environment as a result of atmospheric testing of nuclear weapons also contributes to the background radiation level, although in very small amounts.

The Programmatic SNF&INEL Final EIS presents a summary of the estimated natural background dose by exposure source for residents of the Eastern Snake River Plain (DOE, 1995c). The cumulative annual dose, 351 mrem, is caused largely by the inhalation of radioactive particles formed by the decay of naturally occurring radon.

Sources of radiological emissions at the Idaho National Engineering Laboratory result from operation of research and training reactors, spent nuclear fuel testing and processing, irradiated material and fuel examination, nuclear waste treatment and storage, and depleted uranium armor production. These operations can result in the release of radioactivity to air, either directly (e.g., through stacks or vents) or indirectly (e.g., by re-suspension of radioactivity on contaminated grounds). Concentrations of radionuclides in direct releases are monitored or estimated based on knowledge of the materials used and activities performed. Indirect releases are estimated using engineering calculations that relate surface contamination levels to expected airborne concentrations.

Table 4.7-2 of Appendix B, Volume 1 of the Programmatic SNF&INEL Final EIS provides a summary of the principal types of airborne radioactivity emitted from the Idaho National Engineering Laboratory operations during 1991 (DOE, 1995c). The information summarized provides a perspective on the general nature and magnitude of airborne radiological emissions from current the Idaho National Engineering Laboratory operations. These emissions include the noble gases (argon, krypton, and xenon) and iodine; particulate fission products such as rubidium, strontium, and cesium; radionuclides formed by neutron activation such as tritium (hydrogen-3), carbon-14, and cobalt-60; and very small quantities of heavy elements such as uranium, thorium, plutonium, and their decay products. The emissions listed are considered representative of a maximum baseline year. The radionuclide with the highest emission rate is the noble gas, krypton-85. Most of the krypton-85 emissions result from the chemical reprocessing of spent nuclear fuel at the Idaho Chemical Processing Plant.

3.3.2.5 Ecology

The Idaho National Engineering Laboratory vegetation consists primarily of shrub-steppe vegetation, and is a small fraction of the 45 million ha (112.5 million acres) of this vegetation type found in the Intermountain West. Vegetation communities range from shadscale-steppe vegetation at lower altitudes, through sagebrush and grass-dominated communities, to juniper woodlands along the foothills of the nearby mountains and buttes. Big sagebrush and rabbitbrush are the most common and noticeable shrub species on the Idaho National Engineering Laboratory. Other common shrubs include winterfat, perennial broomweed, black sage, and saltbrush species. Grass species common to the Idaho National Engineering Laboratory include Indian ricegrass, Great Basin wildrye, needle-and-threadgrass, wheatgrasses, bottlebrush squirreltail, and cheatgrass. Common flora species include mustards, summer cypress, and Russian thistle. Fifteen vegetation classes have been identified at the Idaho National Engineering Laboratory. The classes can be grouped into six major types: juniper woodland, native grassland, shrub-steppe, lava, modified, and wetland vegetation types.

The Idaho National Engineering Laboratory supports animal communities typical of Great Basin high desert vegetation and habitats. About 270 vertebrate species have been observed, including 46 mammal, 204 bird, 10 reptile, 2 amphibian, and 9 fish species (Arthur et al., 1984; DOE, 1995c). Thirty-seven mammal species or their signs have been observed at the site. Included are 14 rodents, 4 lagomorphs (rabbits and hares), 6 bats, 6 carnivores, 4 ungulates (hoofed animals), and 1 shrew. A total of 185 bird species were recorded on the site (DOE, 1995c). Other species may be present on the Idaho National Engineering Laboratory since more than 216 bird species have been reported in southeastern Idaho in similar habitats. Of these species, 32 are game birds, 26 are waterfowl, 82 are passerines, and 22 are raptors (for example, hawks, eagles, and owls). Waterfowl use man-made ponds and lagoons at the Idaho National Engineering Laboratory, and the Idaho National Engineering Laboratory is a nesting and wintering area for raptors. Sage grouse and their habitat are found throughout the site. Ten reptile and one amphibian species have been observed on the Idaho National Engineering Laboratory, but only five are common or abundant (DOE, 1995c). These species include western rattlesnake, short-horned lizard, sagebrush lizard, Great Basin spadefoot toad, and western garter snake.

Two Federal endangered and six Federal Category 2 Candidate animal species were identified by the U.S. Fish and Wildlife Service as potentially existing on the Idaho National Engineering Laboratory (DOE, 1995c). Seven additional animal species listed by the State as species-of-special-concern were identified. No Federal-or State-listed plant species were identified as potentially existing on the Idaho National

Engineering Laboratory. The bald eagle and peregrine falcon are Federally listed endangered species. The bald eagle is a winter resident and is locally common in the far north end of the Idaho National Engineering Laboratory, and on the western edge of the site near Howe. Peregrine falcons have been rarely observed in the winter. Neither species is known to nest on the site, and neither species is commonly observed near facilities (Reynolds, 1993).

The Candidate Category 2 species identified by the U.S. Fish and Wildlife Service as potentially existing at the Idaho National Engineering Laboratory include four birds and two mammals. Both the long-billed curlew and the white-faced ibis are uncommon migrants at the Idaho National Engineering Laboratory, associated with aquatic and riparian habitats. Swainson's hawk and ferruginous hawk nest on and migrate through the site. These species are found throughout the site, but are observed more frequently in the juniper woodlands and riparian areas where they nest. The loggerhead shrike, which uses shrub-steppe vegetation to nest, is also found on the Idaho National Engineering Laboratory. Breeding and hibernation caves for Townsend's big-eared bat have been observed on the Idaho National Engineering Laboratory. About six caves are used, the nearest being in excess of 7 km (3 mi) from the nearest facility. The pygmy rabbit is common on the Idaho National Engineering Laboratory (DOE, 1995c). The habitat for this species is grass and shrub communities found throughout much of the Idaho National Engineering Laboratory.

State species-of-special-concern include three aquatic, one raptor, and three bat species. The aquatic bird species are rare migrants (DOE, 1995c). The gray falcon has only been observed a few times. Of the bats, only the California myotis has been observed, and it is rare. Ten plant species identified by other Federal agencies (U.S. Bureau of Land Management and U.S. Forest Service) and the Idaho Native Plant Society as sensitive, rare, or unique are known to occur on the Idaho National Engineering Laboratory (Chowlewa and Henderson, 1984). These species are not, however, protected under State or Federal laws. Most of these species are not found near any the Idaho National Engineering Laboratory facilities, and are uncommon on the Idaho National Engineering Laboratory because they require unique microhabitat conditions.

Potential wetlands are found throughout the Idaho National Engineering Laboratory and cover about 0.25 percent (3,322 ha or 8,206 acres) of the site. The wetlands are associated primarily with drainages, although some wetlands are associated with natural playas, man-made ponds, and drainage ditches. More than 70 percent of the potential wetlands are identified as palustrine (marsh lands). These wetlands are found mainly near the Big Lost River and its spreading areas and playas, the Birch Creek Playa, and an area to the north and in the general vicinity of Argonne National Laboratory-West. Potential lacustrine wetlands are found in the central and south central portions of the site. Potential riverine wetlands were mapped by the U.S. Fish and Wildlife Service near the Big Lost River, Birch Creek, and Big Lost River Playas. Man-made wetlands include industrial waste and sewage treatment ponds, borrow pits, and gravel pits associated with site facilities. Limited riparian communities with large trees are found along the drainages of the Big Lost River, and very limited riparian habitat is located on Birch Creek.

3.3.2.6 Land Use

The Idaho National Engineering Laboratory encompasses 231,049 ha (570,934 acres). Only about two percent of the land (4,614 ha or 11,400 acres) is used for facilities and operations supporting energy research and development and waste management. In addition to industrial and support land uses associated with each of the facility areas at the Idaho National Engineering Laboratory, other land uses exist within the Idaho National Engineering Laboratory site boundaries.

Approximately five percent of the total the Idaho National Engineering Laboratory site area, or 13,349 ha (33,000 acres), is devoted to public road, utility, and railway rights-of-way crossing the site (DOE, 1995c). Rights-of-way at the Idaho National Engineering Laboratory are granted and administered by the U.S. Department of the Interior and the Bureau of Land Management.

More than half of the Idaho National Engineering Laboratory is used for grazing. The actual amount of the Idaho National Engineering Laboratory land used for grazing varies from year to year, but is usually between 121,410 and 141,645 ha (300,000 and 350,000 acres). In 1992 for example, 121,853 ha (301,094 acres) were grazed (DOE, 1995c). Grazing is not allowed within 3.2 km (2 mi) of any nuclear facility, and dairy cattle are not permitted on the site to avoid the possibility of milk contamination by long-lived radionuclides. The U.S. Sheep Experiment Station, located approximately 42.6 km (26.5 mi) northeast of the site, uses a 364 ha (900 acre) portion of the Idaho National Engineering Laboratory for a winter feed lot for approximately 5,000 sheep.

The Idaho National Engineering Laboratory also supports periodic uses associated with onsite resources. Two sites within the Idaho National Engineering Laboratory site boundary are listed in the National Register of Historic Places: Experimental Breeder Reactor-I, designated in 1966, and Goodale's Cutoff, a portion of the Oregon Trail that crosses the southwest corner of the Idaho National Engineering Laboratory, designated in 1974. In addition to public tours, the Idaho National Engineering Laboratory occasionally supports controlled hunting within the site boundaries. Each year the Idaho Department of Fish and Game and DOE jointly determine whether or not to allow controlled hunts of wild game populations living on the Idaho National Engineering Laboratory property (DOE, 1995c).

Several uses associated with the National Environmental Research Park designation occur at the Idaho National Engineering Laboratory. The Idaho National Engineering Laboratory's cool desert ecosystem provides a controlled outdoor laboratory where scientists can study changes to the natural environment caused by human activities.

3.3.2.7 Noise

Sources of man-made noise at the Idaho National Engineering Laboratory include noise from operation and construction activities; bus, car, and truck traffic; aircraft; security force training exercises; and the Idaho National Engineering Laboratory railroad. Previous studies have assessed noise impacts of existing the Idaho National Engineering Laboratory operational activities (Abbott et al., 1990). These studies concluded that because of the remote location of the Idaho National Engineering Laboratory, there are no known noise conditions associated with existing onsite operations that adversely affect individuals at offsite locations. Studies of the noise effects on wildlife at the Kennedy Space Center show that even very high noise levels have little significance on wildlife productivity (Leonard, 1993).

3.3.2.8 Transportation

Roads are the primary access to and from the site, and workers are transported primarily by bus and private vehicles. Commercial shipments are transported by road and air, and some bulk materials are transported by rail. Waste is transported by road and rail. The existing regional highway system and site roadways are shown in Figure 3-54. To maintain a supply of goods and services and to transport workers to the Idaho National Engineering Laboratory, an onsite road system of approximately 145 km (90 mi) of paved surface has been developed. About 29 km (18 mi) of this network are considered service roads and are closed to the public. In addition to the site facilities, DOE owns or leases office and technical buildings throughout Idaho Falls for approximately 4,000 DOE and DOE contractor personnel to administer and support work at the Idaho National Engineering Laboratory. DOE shuttle vans provide

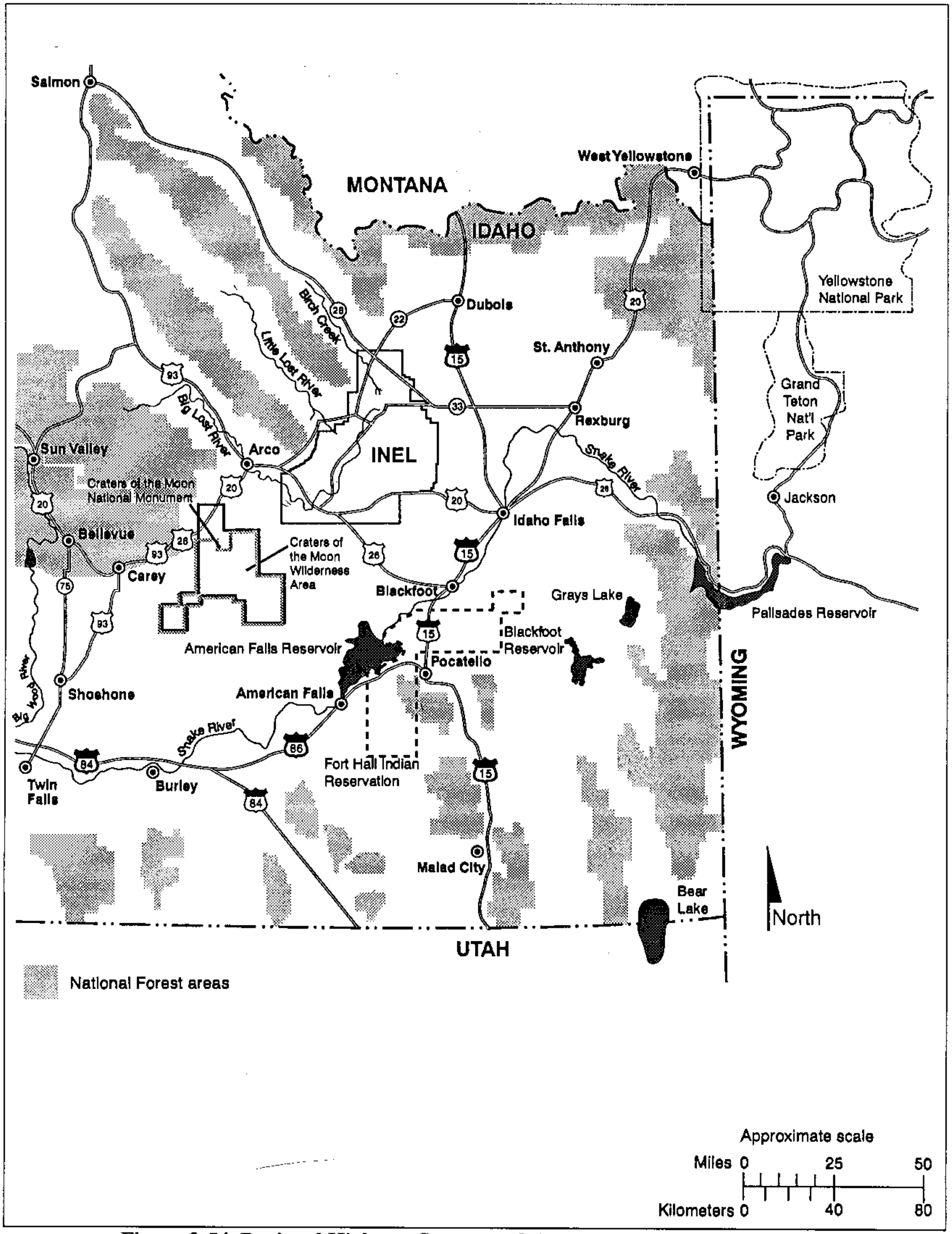


Figure 3-54 Regional Highway System and the Idaho National Engineering Laboratory Site Roadways

hourly transport between in-town facilities. Four major modes of transit use the regional highways, community streets, and site roads to transport people and commodities: DOE buses and shuttle vans, DOE motor pool vehicles, commercial trucks, and private automobiles.

Idaho Falls receives railroad freight service from Butte, Montana to the north, and from Pocatello, and Salt Lake City, Utah to the south via Union Pacific. The Union Pacific Railroad's Mackay Branch, which crosses the southern portion of the Idaho National Engineering Laboratory, provides rail service to the Idaho National Engineering Laboratory. This branch connects with a DOE-owned spur line at Scoville Siding, and then links with developed areas within the Idaho National Engineering Laboratory. Rail shipments to and from the Idaho National Engineering Laboratory are usually limited to bulk commodities, spent nuclear fuel, and radioactive materials.

One major airline carrier, Delta Airlines, provides Idaho Falls with jet aircraft passenger and cargo service. Two other air carriers, Horizon and Skywest, provide commuter service. Daily service includes flights to and from Boise and Salt Lake City. Landings in Idaho Falls for 1991 and 1992 totalled 5,367 and 5,578, respectively. The combined number of passengers leaving and arriving at Idaho Falls and Pocatello for 1991 and 1992 was 282,185 and 285,047, respectively. Non-DOE air traffic over the Idaho National Engineering Laboratory is restricted, and non-DOE aircraft are not permitted to use the site.

From 1987 through 1992, the average motor vehicle accident rate was 0.94 accident per million km (1.5 accidents per million mi) for Idaho National Engineering Laboratory vehicles, which compares with an accident rate of 1.5 accidents per million km (2.4 accidents per million mi) for all DOE complex vehicles and 8 accidents per million km (12.8 accidents per million mi) nationwide for all motor vehicles. There are no recorded rail or air accidents associated with the Idaho National Engineering Laboratory and, to date, no fatal air traffic accidents have involved flights through either the Idaho Falls or Pocatello airports (DOE, 1995c). Spent nuclear fuel and radioactive, hazardous, industrial, municipal, and recyclable wastes are transported at the Idaho National Engineering Laboratory.

3.3.2.9 Socioeconomics

In general, population growth in the region has mirrored population growth in Idaho as a whole over the past 30 years (Figure 3-55). Although growth was not evenly distributed among the counties, total regional population increased by 47 percent between the years 1960 to 1990, which is comparable to a Statewide growth rate of 51 percent. During this period, Madison County experienced the most rapid growth in the region, its population increasing by more than 150 percent. By contrast, the two smallest counties in the region, Butte and Clark, actually lost population between the years 1960 and 1990. The largest increases in the population of Idaho Falls occurred during the 1950's, which can be partially explained by the formation and the development of the National Reactor Testing Station, which evolved into the Idaho National Engineering Laboratory. By contrast, Pocatello grew primarily in the 1940's and the 1960's. Idaho Falls has continued to grow, increasing in population by 11 percent from the years 1980 to 1990. During the same period, however, Pocatello's population declined by 0.7 percent.

Historically, the economy of the seven-county region relied predominantly on natural resource use and extraction. To this day, farming, ranching, and mining remain important components of the regional economy. Almost all manufacturing in the region is a form of food processing, and mining or mineral processing. Idaho Falls is a regional retail and service center for southeastern Idaho. Similarly, Pocatello has evolved into an important processing and distribution center for the surrounding agricultural areas, and is a regional center for higher education. Retail trade and educational services are the two largest regional employment categories, providing 17.6 and 11.4 percent of the total employment in the region. Tourism is also considered to be an important component of the regional economy. The economy of the seven-county

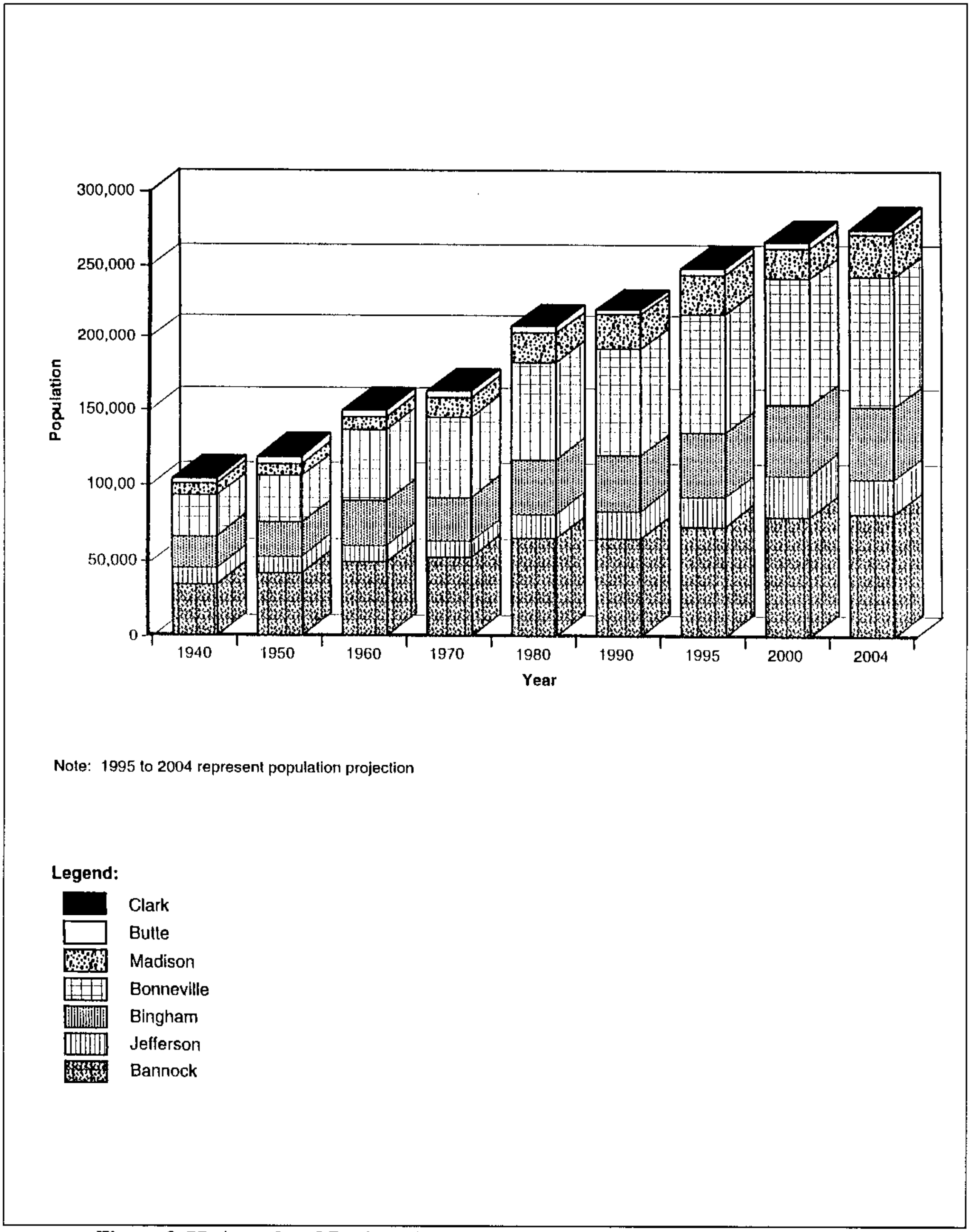


Figure 3-55 Actual and Projected Total Population for the Seven-County Region Surrounding the Idaho National Engineering Laboratory for the Years 1940 through 2004

region, however, currently revolves around agriculture and the Idaho National Engineering Laboratory (the single largest employer in the seven-county region). As of January 1991, DOE and its contractors employed more than 12,425 persons, with an estimated annual payroll of $440 million. Annual funding for the Idaho National Engineering Laboratory in 1991 was more than $1.1 billion.

As of January 1992, 12,803 contractor and Government personnel were employed at the Idaho National Engineering Laboratory, representing about 12 percent of the total available jobs in the seven-county region. Total employment at the Idaho National Engineering Laboratory has increased by more than 23 percent in the past 4 years. The Idaho National Engineering Laboratory has a large influence on both the regional economy and the economy of Idaho. During Fiscal Year 1992, total expenditures at the Idaho National Engineering Laboratory directly and indirectly supported 36,395 jobs, and generated an estimated $945 million in total earnings within the seven-county region.

Bonneville and Bannock Counties are the focal points of the housing market. These two counties provide 67 percent of the total housing in the region. A shortage of single-family housing presently exists near Idaho Falls because construction has not kept up with increasing demand. Total population living within the 80 km (50 mi) radius around the site is about 176,311. The ethnic composition of this population is presented in Figure 3-56. The number of low-income households within the same area of influence is presented in Figure 3-57. Approximately 40 percent of these households are low income families.

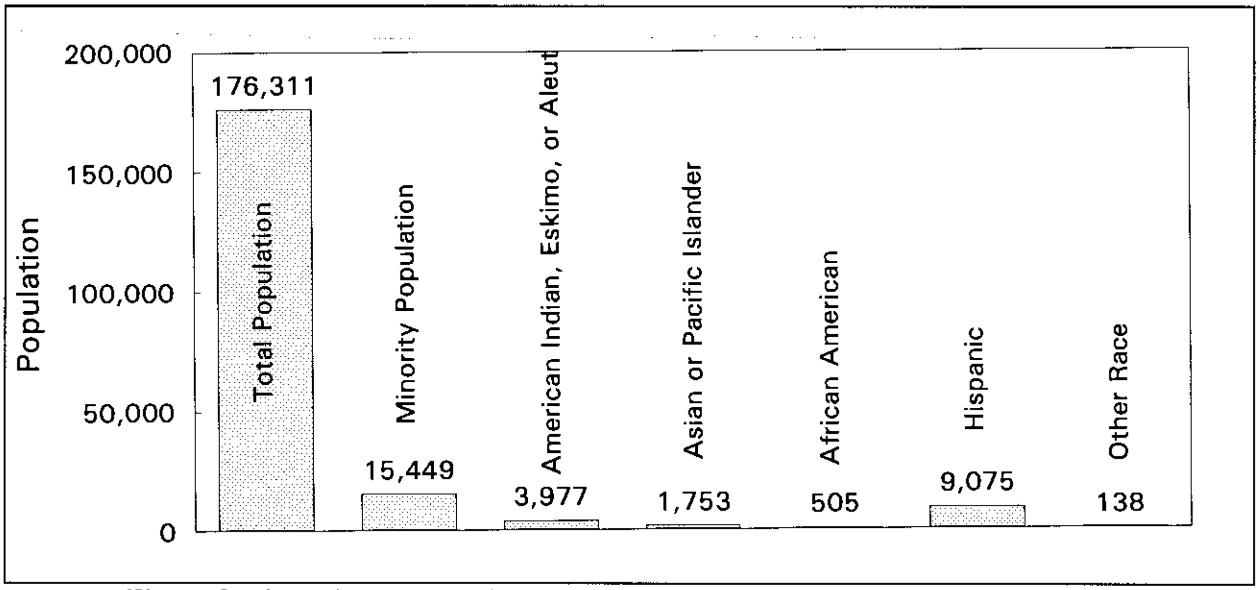


Figure 3-56 Racial and Ethnic Composition of the Minority Population Residing within 80 km (50 mi) of the Idaho National Engineering Laboratory

Seventeen public school districts provide educational services for 56,899 school-aged children within the seven-county region. Most of the public schools in the seven-county region operate at levels at or above the design capacity of their classroom facilities.

There are 18 fire districts in the seven-county area surrounding the Idaho National Engineering Laboratory. These 18 districts operate a total of 30 fire stations staffed by 179 paid and 313 volunteer firefighters. Bingham, Bonneville, Butte, Clark, and Jefferson counties, which surround the Idaho National Engineering Laboratory, have developed emergency plans to be implemented in the event of a radiological or hazardous materials emergency. Eight hospitals serve the seven-county region. The Eastern Idaho Regional Medical Center in Idaho Falls, with 311 beds, is the largest hospital in the region. Occupancy rates range from 22.0 to 89.2 percent in the region. Law enforcement services in the

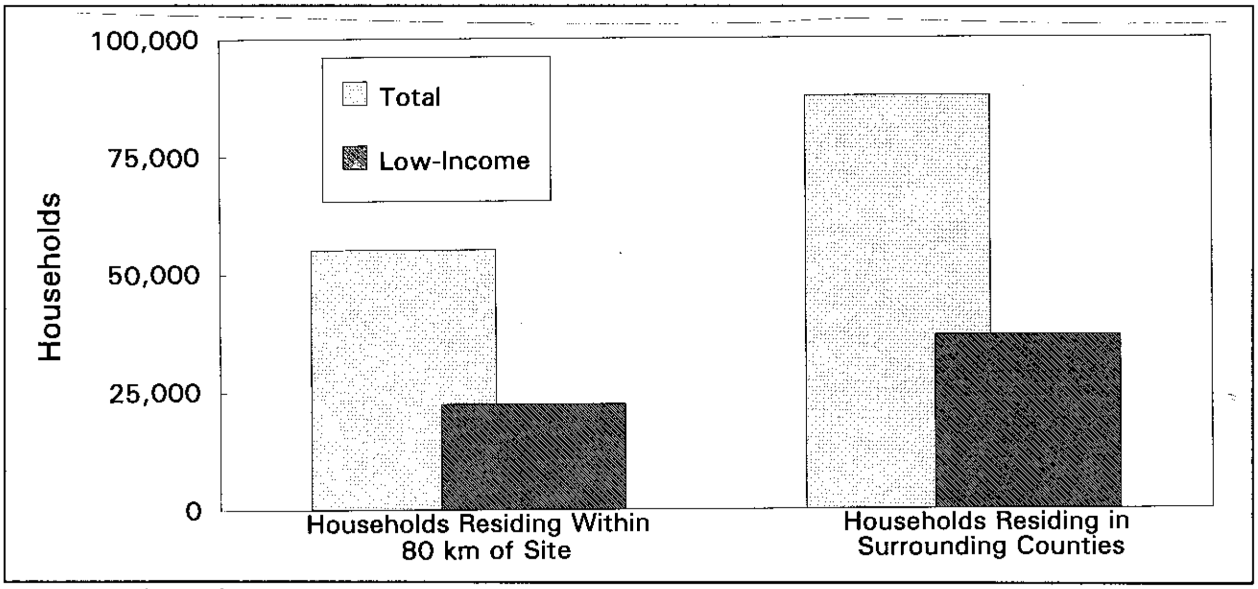


Figure 3-57 Low-Income Households Residing within 80 km (50 mi) of the Idaho National Engineering Laboratory

seven-county area are provided by sheriff's offices in each county, 12 city police departments, and the Idaho State Police. Clark County has the highest ratio of law enforcement personnel per 1,000 people (6.3 per 1,000), and Butte County has the lowest ratio (1.3 per 1,000).

3.3.2.10 Historical, Archaeological, and Cultural Resources

In the course of more than 100 cultural resource surveys, 1,506 cultural resources have been discovered and recorded within the Idaho National Engineering Laboratory boundaries (DOE, 1995c), not including architectural properties directly associated with the creation and operation of the Idaho National Engineering Laboratory. Only four percent of the Idaho National Engineering Laboratory has been surveyed, however, and most surveys were conducted near major facility areas. The 1,506 resources recorded at the Idaho National Engineering Laboratory through 1992 include 688 prehistoric sites, 38 historic sites, 753 prehistoric isolated finds, and 27 historic isolated finds (DOE, 1995c; Gilbert and Ringe, 1993). Until formal evaluations can be performed, all of the cultural sites are considered to be potentially eligible for the National Register of Historic Places. Only Goodale's Cutoff, part of the Northern Alternate of the Oregon Trail, is listed on the National Register of Historic Places. All of the isolated find artifacts have been categorized as unlikely to meet eligibility requirements (Yohe, 1993).

Most of the 688 prehistoric archaeological sites located to date at the Idaho National Engineering Laboratory are classified as lithic scatters. These sites consist of more than 10 stone artifacts, but lack evidence of other categories of cultural materials or features. At 12 percent of these sites, limited subsurface excavations have been conducted. Prehistoric cultural resources vary in density and type across the Idaho National Engineering Laboratory, but appear to be associated with definable physical features of the land. Areas classified as having very high sensitivity have been identified along the Big Lost River, atop buttes, and within craters and caves. The southernmost part of the Lemhi Mountains, the Lake Terreton basin, and a 2,800 m (9,200 ft) wide zone along the edge of lava fields are classified as high-sensitivity areas. With the exception of the Central Facilities Area and Experimental Breeder Reactor-I, which are located in high-sensitivity zones, all developed Idaho National Engineering Laboratory facility areas are located in or on the edge of low or medium sensitivity areas.

As of March, 1993, 38 historic sites and 27 historic isolated finds had been recorded on the Idaho National Engineering Laboratory (Gilbert and Ringe, 1993). These resources include small homesteads and irrigation canals, temporary campsites of sheep and cattle drivers, stage and wagon roads, and some mining-related features.

In most instances, a property must be at least 50 years old to be considered for inclusion in the National Register of Historic Places, unless it meets certain criteria. The unique nature of existing DOE site facilities at the Idaho National Engineering Laboratory and their historic role in nuclear research and development make them eligible for special consideration regardless of their age. Two buildings dating from 1942 appear on a partial inventory of potentially significant historic the Idaho National Engineering Laboratory facilities (Braun et al., 1993). The Experimental Breeder Reactor-I is listed in the National Register of Historic Places, and is designated as a National Historic Landmark. Although the Experimental Breeder Reactor-I is the only nuclear property less than 50 years old that has been formally listed in the National Register, other DOE facilities eligible for listing have been identified, including the Auxiliary Reactor Areas I, II, III; Boiling Water Reactor Experiment V; Materials Test Reactor; Engineering Test Reactor; and the Test Area North hangar (Braun et al., 1993). The Idaho State Historic Preservation Office believes that most major structures related to nuclear research at the Idaho National Engineering Laboratory are probably eligible for the National Register (Yohe, 1993).

As of July 1993, several specific types of locations have been identified as associated with traditional religious practices. The most significant of these locations are the East and Middle Buttes located in the southern section of the Idaho National Engineering Laboratory, and more than 20 lava tube and blister cave sites, which are considered sacred by the Shoshone-Bannock Tribes. The Tribes also consider archaeological sites to be sensitive resources, especially rock art sites. The entire the Idaho National Engineering Laboratory falls within the traditional territory of the Shoshone-Bannock Tribes. These areas of concern are likely to be located within very high sensitivity zones identified for presence of prehistoric archaeological resources, such as the Big Lost River, Birch Creek, buttes, craters, caves, lava edge zones, Lemhi Mountains, and the Lake Terreton basin

3.3.3 Description of the Affected Environment at the Hanford Site

This section describes the potentially affected environment of the Hanford Site. The location of the Hanford Site is shown in Figure 3-58.

3.3.3.1 Geology

The region of the Pacific Northwest that contains the Hanford Site lies within the Columbia Intermontane physiographic province, which is bordered on the north and east by the Rocky Mountains, and on the west by the Cascade Range (Figure 3-58). The province has been a topographic and structural depression since the early Miocene period, and is subdivided into smaller physiographic units. The dominant geologic characteristics of this province are the thick accumulations of basaltic lava flows extending laterally from central Washington eastward into Idaho, and southward into Oregon (Tallman et al., 1979). The ancient basalt surface has been subsequently modified by tectonism, volcanism, weathering, and erosion.

The Columbia Intermontane Province is divided into four subprovinces. The Hanford Site is contained within the Columbia Basin subprovince, which contains most of the Columbia River Basalt Group. The Columbia Basin subprovince is further divided into six physiographic sections, with the Hanford Site located in parts of the Yakima Folds and the Central Plains sections. Much of the Columbia Basin subprovince was affected by preglacial cataclysmic flooding associated with the sudden release of water from glacial Lake Missoula. Cataclysmic floods were responsible for much of the present morphology of

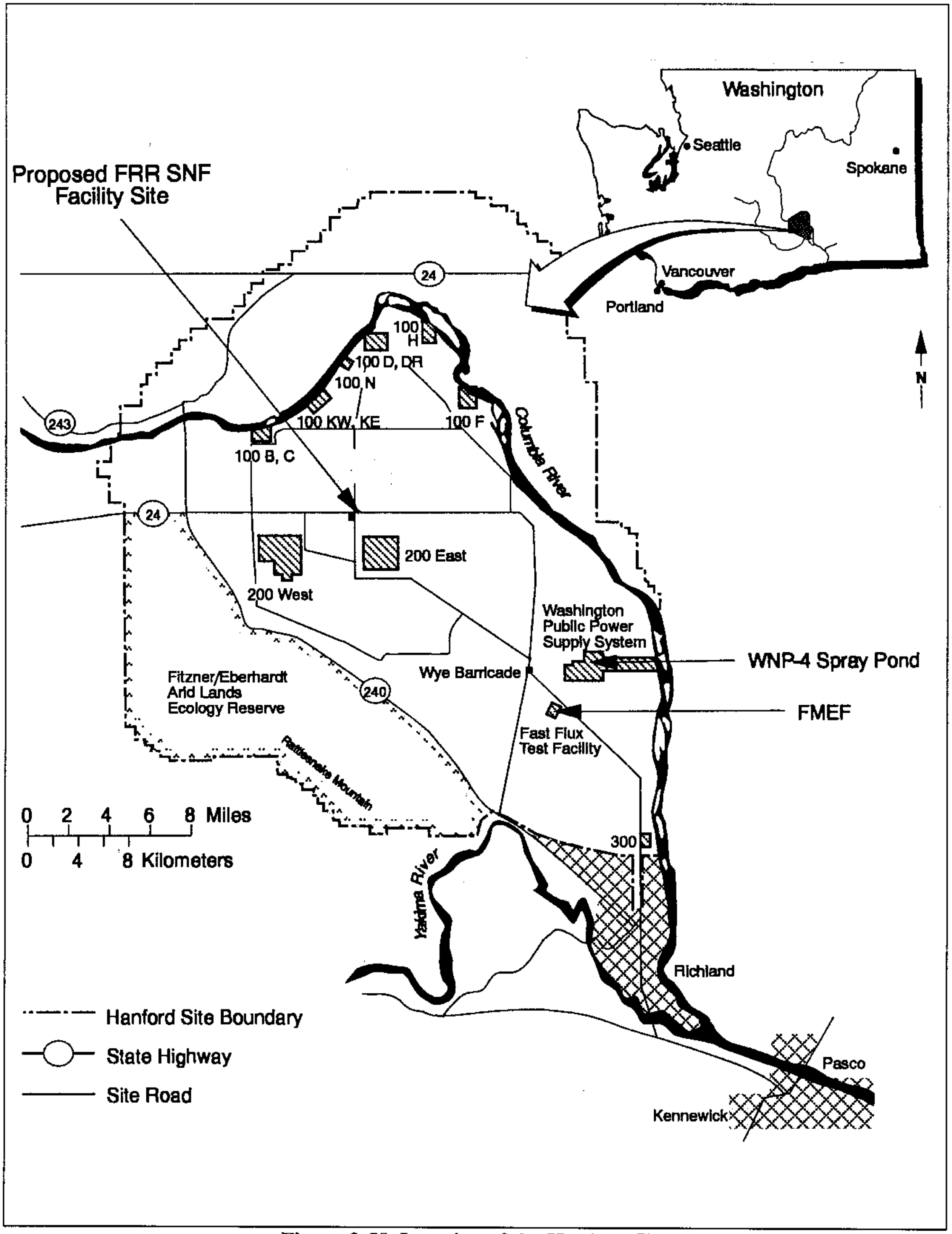


Figure 3-58 Location of the Hanford Site

the Channeled Scabland and Central Plains section. Fluvial and lacustrine processes associated with the ancestral Columbia River system have been active since the late Miocene. Sedimentary deposits indicate that deposition was continuous from about 10.5 million years ago until about 3.5 million years ago (DOE, 1988c). Quaternary volcanism has been limited to the extreme western margin of the Columbia Basin subprovince, and is associated with the Cascade Range Province.

The Hanford Site is located within the Pasco Basin, which was formed by the deformation of the lava flows into broad structural and topographic basins. The Pasco Basin is defined by anticlinal structures of basaltic rock known as the Saddle Mountains to the north, the Umtanum Ridge, Yakima Ridge, and Rattlesnake Hills to the west, and a series of doubly plunging anticlines merging with the Horse Heaven Hills to the south.

Most known faults within the region are associated with anticlinal fold axes, and were probably formed concurrently with the folding (DOE, 1986a). Existing known faults within the Hanford Site area include tear faults with lengths of up to 3 km (1.9 mi) on Gable Mountain, and the Rattlesnake-Wallula alignment. Strike-slip faults have not been observed crosscutting the Pasco Basin. Structures within the Hanford Site have shown the greatest deformation along the hinge area of the anticlinal ridges, decreasing with distance from that area (i.e., 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, 1986a).

Fifteen different soil types present on the Hanford Site have been listed and described (Hajek, 1966). The soil types vary from sand to silty and sandy loam.

3.3.3.2 Seismology and Volcanology

Seismicity of the Columbia Plateau, as determined by the rate of earthquakes per area and the historical magnitude of these events, is relatively low compared to other regions of the Pacific Northwest, the Puget Sound area, and western Montana/eastern Idaho. Figure 3-59 shows the locations of all earthquakes that occurred in the Columbia Plateau from 1850 to 1969 with Modified Mercalli Intensity of 5 or greater. Swarms of small, shallow earthquakes lasting from several weeks to months, and clustered in an area 5 to 10 km (3 to 6 mi) in lateral dimension are the predominant seismic events. Earthquake swarms may contain from four to more than 100 earthquakes of magnitude 1.0 to 3.5. Detailed locations of swarm earthquakes indicate that the events occur on fault planes of variable orientation and not on a single throughgoing fault plane (DOE, 1995c).

Shallow earthquake swarm activity in the central Columbia Plateau is concentrated principally north and east of the Hanford Site. Here, earthquakes of magnitude greater than 3.0 occur, with the largest recorded (in 1973) having magnitude 4.4 north of the Hanford Site. Deeper earthquakes occur in the central Columbia Plateau, although at much lower frequencies than the shallower swarm events. Deep seismic activity generally occurs randomly, and is not associated with known geologic structures or with patterns of shallow seismicity. Earthquake focal 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). The earthquake focal mechanisms in 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).

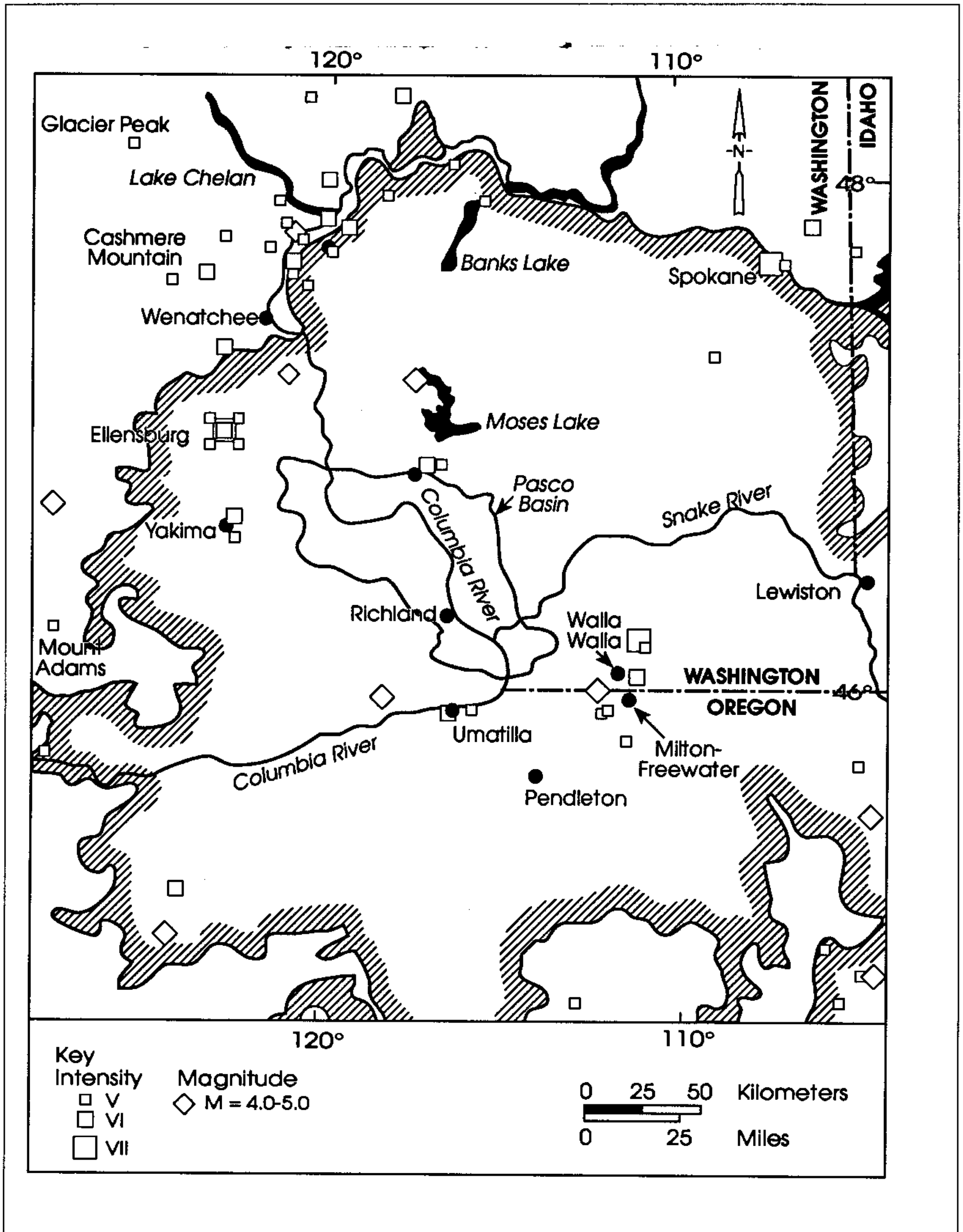


Figure 3-59 Historical Seismicity of the Columbia Plateau (DOE, 1995c)

There are several major volcanoes in the Cascade Range west of the Hanford Site. The nearest volcano is Mount Adams, about 165 km (103 mi) from the Hanford Site, and the most active is Mount St. Helens, approximately 220 km (137 mi) west-southwest from the Hanford Site.

3.3.3.3 Hydrology

The major geologic units of the Hanford Site are, in ascending order: basement rocks of unknown origin and composition, the Columbia River Basalt Group with interbedded sediments of the Ellensburg Formation, the Ringold Formation, the Plio-Pleistocene unit, and the Hanford formation.

3.3.3.3.1 Surface Water

The Hanford Site occupies approximately 33 percent 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 are generally associated with fuel and waste processing activities (Figure 3-60). There is no significant surface water flow from the Hanford Site to nearby rivers.

Cold Creek and its tributary, Dry Creek, are ephemeral streams within the Yakima River drainage system along the southern boundary of the Hanford Site. Rattlesnake Springs, located on the western part of the Hanford Site, forms a small surface stream that flows for about 3 km (1.9 mi) before disappearing into the ground.

Normal river elevations within the Hanford Site range from 120 m (396 ft) above mean sea level when the river enters the Hanford Site near Vernita, to 104 m (343 ft) where it leaves the Hanford Site near the 300 Area.

Large Columbia River floods have occurred in the past (DOE, 1986a), but the likelihood of recurrence of large-scale flooding has been reduced by the construction of several flood control/water storage dams upstream from the Hanford Site. The maximum historical flood on record occurred June 7, 1894, with a peak discharge at the Hanford Site of 21,000 m^3 per sec (741,600 ft^3 per sec) (Figure 3-61). The largest recent flood took place in 1948, with an observed peak discharge of 20,000 m^3 per sec (706,300 ft^3 per sec) at the Hanford Site. The probability of flooding at the magnitude of the 1894 and 1948 floods has been greatly lowered because of upstream regulation by dams (Figure 3-60). There have been fewer than 20 major floods on the Yakima River since 1862 (DOE, 1986a).

The probable maximum flood for the Columbia River below Priest Rapids Dam has been calculated to be 40,000 m^3 per sec (1,412,600 ft^3 per sec). This flood would inundate the 100 Areas located adjacent to the Columbia River, but the central portion of the Hanford Site would remain unaffected (DOE, 1986a).

The U.S. Army Corps of Engineers evaluated a number of scenarios on the effects of failures of Grand Coulee Dam, assuming flow conditions on the order of 11,000 m^3 per sec (388,500 ft^3 per sec). The discharge resulting from a 50 percent breach at the outfall of Grand Coulee Dam was determined to be 600,000 m^3 per sec (21,188,800 ft^3 per sec). In addition to the areas inundated by the probable maximum flood, the remainder of the 100 Areas, the 300 Area, and nearly all of Richland, WA, would be flooded (DOE, 1986a).

Surface Water Quality: The Washington State Department of Ecology classifies the Columbia River as Class A (excellent) between Grand Coulee Dam and the mouth of the river near Astoria, OR (DOE, 1986a). The Class A designation requires that industrial uses of this water be compatible with other uses, including drinking water, wildlife, and recreation.

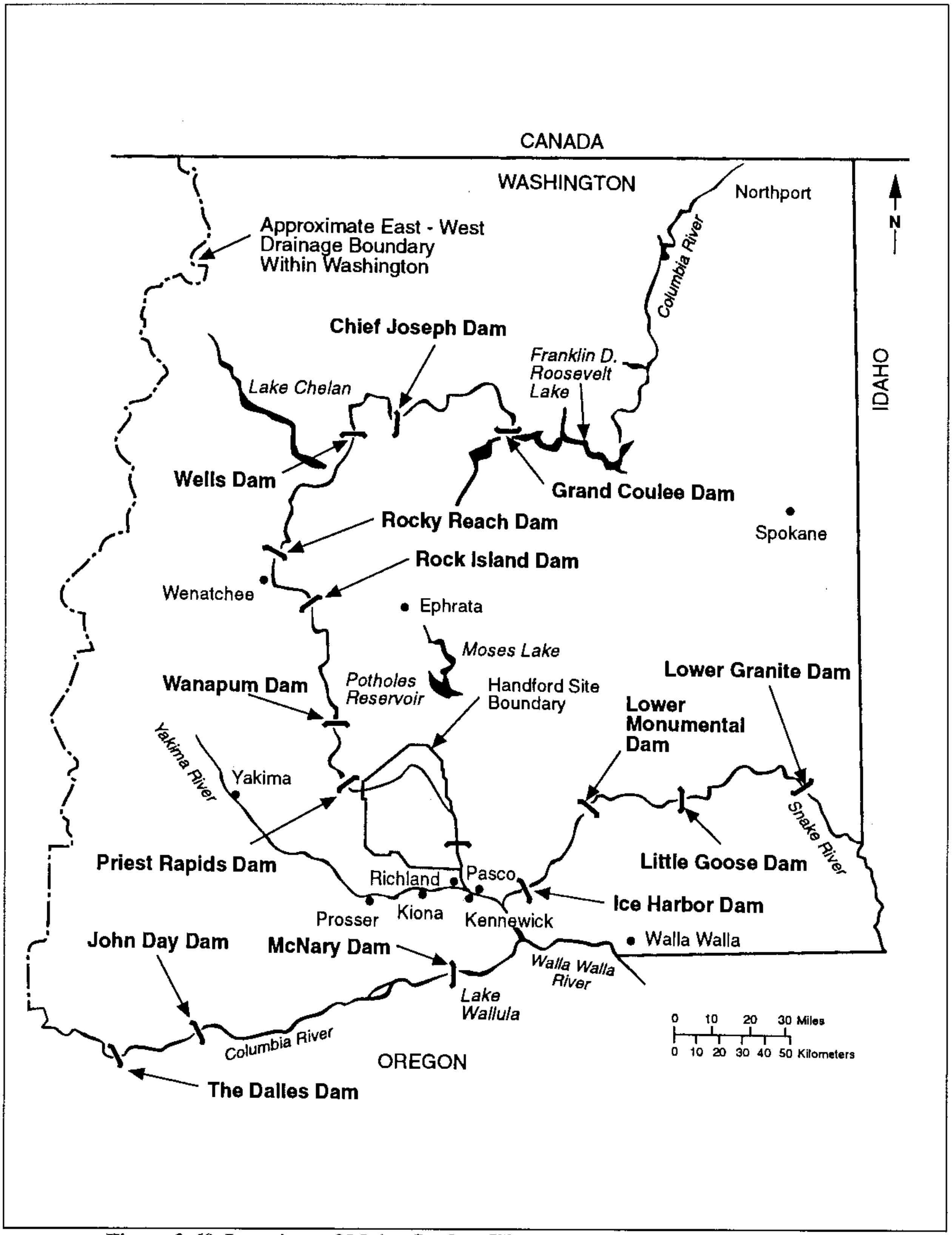


Figure 3-60 Locations of Major Surface Water Resources and Principal Dams within the Columbia Plateau

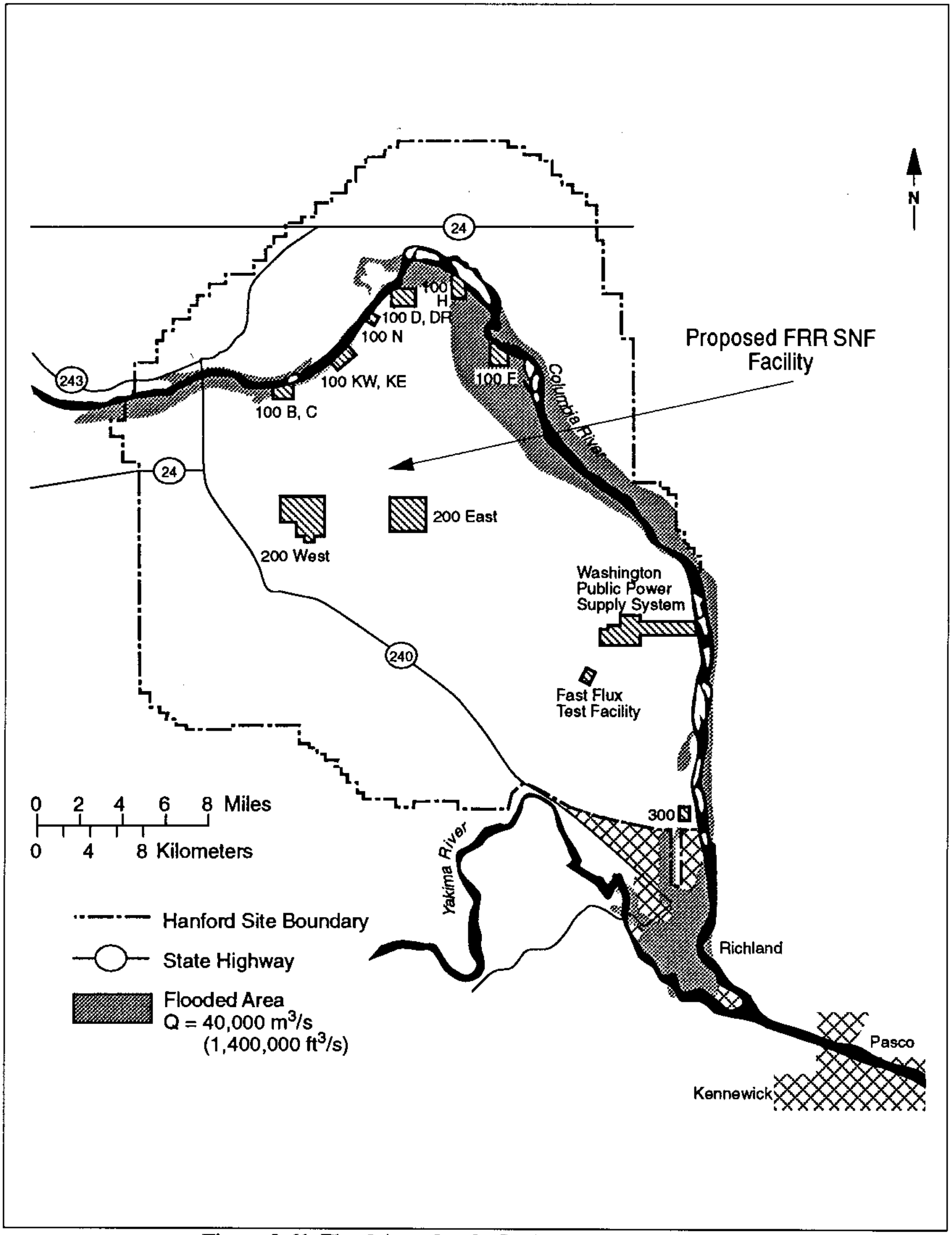


Figure 3-61 Flood Area for the Probable Maximum Flood

Radiological monitoring shows low levels of radionuclides in samples of Columbia River water. Hydrogen-3 (tritium), iodine-129, and uranium are found in slightly higher concentrations downstream from the Hanford Site than upstream (DOE, 1995c), but were well below concentration guidelines established by DOE and U.S. Environmental Protection Agency drinking water standards. Cobalt-60 and iodine-131 were not consistently found in measurable quantities during 1987 samples of Columbia River water from Priest Rapids Dam, the 300-Area water intake, or the Richland City pumphouse. The average annual strontium-90 concentrations were essentially the same at Priest Rapids Dam and the Richland Pumphouse for 1987, and were well below the State of Washington and U.S. Environmental Protection Agency drinking water standards.

3.3.3.3.2 Groundwater

Groundwater under the Hanford Site occurs in unconfined and confined conditions. The unconfined aquifer is contained within the glaciofluvial sands and gravel and within the Ringold Formation. The bottom of the aquifer is the basalt surface or, in some areas, the clay zones of the lower member of the Ringold Formation. 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.

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 River and Columbia River.

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, there is also a component of groundwater flow 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. Groundwater discharge from the shallow basalt is probably to the overlying unconfined aquifer and the Columbia River. The discharge area(s) for the deep groundwater is currently uncertain, but flow is believed to be generally southeastward with discharge speculated to be south of the Hanford Site (DOE, 1986a).

Groundwater Quality: The groundwater composition is that of a dilute (less than or approximately 350 mg per L total dissolved solids) calcium-bicarbonate chemical type. Other principal chemical constituents include sulfate, silica, magnesium, and nitrate (the latter contributed through the disposal of chemical reprocessing waters).

Contamination of the groundwater is caused by releases from various liquid-waste disposal facilities. Nitrate and tritium contamination has migrated away from these sites in a general west-to-east direction. Some longer-lived radionuclides such as strontium-90 and cesium-137 have reached the groundwater, primarily through liquid-waste disposal cribs. Small quantities of longer-lived radionuclides have reached the water table via a failed groundwater monitoring well casing and through reverse well injection, a disposal practice which was discontinued at the Hanford Site in 1947 (DOE, 1995c).

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. Overall, waters of the shallow basalts are of a sodium-bicarbonate chemical type, while those of the deep basalts are of a sodium-chloride chemical type (DOE, 1986a). Iodine-129 and tritium have been detected in confined groundwater in the Saddle Mountains Basalt beneath the Hanford Site (DOE, 1986a).

3.3.3.4 Meteorology

The Hanford Site is located in a semi-arid region of southeastern Washington State. A summary of the following data (through 1980) has been published (Stone et al., 1983).

Wind: Prevailing wind directions on the 200-Area Plateau are from the northwest in all months of the year. Monthly and annual joint frequency distributions of wind direction versus wind speed for the Hanford Meteorological Station have been recorded (Stone et al., 1983). Monthly average wind speeds are lowest during the winter months, averaging 10 to 11 km per hr (6.2 to 6.8 mph), and highest during the summer, averaging 14 to 16 km per hr (8.7 to 9.9 mph). High winds are also associated with thunderstorms. The average occurrence of thunderstorms is 10 per year. Although thunderstorms are most frequent during the summer, they have occurred in each month (DOE, 1995c). It is estimated that the probability of a tornado striking a point at the Hanford Site is 0.0000096 per year, or less than one in one-hundred thousand.

Temperature and Humidity: Diurnal and monthly averages and extremes of temperature, dew point, and humidity have been recorded (Stone et al., 1983). Ranges of daily maximum and minimum temperatures vary from normal maxima of 2°C (35.6°F) in early January to 35°C (95°F) in late July. The record maximum temperature is 46°C (114.8F), and the record minimum temperature is -32.8°C (-27.0°F). 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.

Precipitation: Average annual precipitation at the Hanford Meteorological Station is 16 cm (6.3 in). Most of the precipitation occurs during the winter, with nearly half of the annual amount occurring in the months of November through February. Winter monthly average snowfall ranges from 0.8 cm (0.3 in) in March to 13.5 cm (5.3 in) in January. Snowfall accounts for about 38 percent of all precipitation during the months of December through February.

Atmospheric Dispersion: Good dispersion conditions associated with neutral and unstable stratification exist about 57 percent of the time during the summer.

Air Quality: Air quality in the vicinity of the Hanford Site is generally classified as quite good. Wind-eroded dust, resulting from plowed fields and arid terrain with sparse vegetation, is an occasional problem in the area. The atmospheric conditions that produce the dust are favorable to pollutant transport and diffusion.

3.3.3.5 Ecology

The Hanford Site is made up of relatively large, undisturbed (1,450 km^2, about 560 mi^2) expanses of shrub-steppe desert that contain numerous plant and animal species suited to the region's semi-arid environment. The Hanford Site consists of mostly undeveloped land, with facilities only occupying about 6 percent of the total available area. Most of the Hanford Site has not experienced tillage or livestock grazing since the early 1940's.

The Hanford Site vegetation has been characterized as a shrub-steppe (DOE, 1995c) with relatively low productivity. In the early 1800s, the dominant plant was the big sagebrush with an understory of perennial bunchgrasses, but with the advent of livestock and crop raising, the natural vegetation was overtaken by what is today the dominant plant, cheatgrass. Some planted trees exist and 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). Today, the vegetation picture at the Hanford Site consists of eight major kinds of plant communities: sagebrush/bluebunch wheatgrass, sagebrush/cheatgrass or sagebrush/Sandberg's bluegrass, sagebrush-bitterbrush/cheatgrass, greasewood/cheatgrass-saltgrass, winterfat/Sandberg's bluegrass, thyme buckwheat/Sandberg's bluegrass, cheatgrass-tumble mustard, willow or riparian, spiny hopsage, and sand dunes. More than 600 species of plants have been identified at the Hanford Site (Sackschewsky et al., 1992). A distribution of the dominant plant communities is shown in Figure 3-62.

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 along with other species, are important in the food web of the local birds and mammals. Approximately 12 species of amphibians and reptiles are known to exist on the Hanford Site (Fitzner and Gray, 1991). The most abundant reptile is the side-blotched lizard, although short-horned and sagebrush lizards are also seen frequently. Also common are gopher snakes, yellow-bellied racers, the Pacific rattlesnake, toads, and frogs.

More than 125 species of birds have been identified on the Hanford Site (Rogers and Rickard, 1977). The horned lark and western meadowlark are the most abundant nesting birds in the shrub-steppe. The Hanford Site supports populations of chukar partridge, gray partridge, and sage grouse, with the greatest concentration in the Rattlesnake Hills. Wastewater ponds at the Hanford Site are important habitats for songbirds, shore birds, ducks, and geese. The most important waterfowl is the Canada goose. Hawks and owls use the Hanford Site as a refuge, especially during nesting (DOE, 1995c).

Approximately 39 species of mammals have been identified on the Hanford Site (Fitzner and Gray, 1991). Most are small and nocturnal. Of this group, the Great Basin pocket mouse is the most abundant, and others include the deer mouse, Townsend ground squirrel, pocket gopher, harvest mouse, Norway rat, sagebrush vole, grasshopper mouse, vagrant, Leasts chipmunk, and Merriam vole. Larger mammals include the mule deer and elk. The largest vertebrate predator inhabiting the Hanford Site is the coyote. A herd of wild, free-roaming elk is centered almost entirely on the Arid Lands Ecology Reserve, a part of the Hanford Site established as an environmental research study area in 1968.

The Columbia River is the dominant aquatic ecosystem, and the river supports a large and diverse community of plankton, benthic invertebrates, fish, and other communities. Plankton populations in the Hanford reach are influenced by communities that develop in the reservoir of upstream dams, with phytoplankton and zooplankton populations being largely transient, flowing from one reservoir to another. Phytoplankton species include diatoms, golden or yellow-brown algae, red algae, and dinoflagellates. Zooplankton populations in the Hanford reach of the Columbia are generally sparse. All major freshwater benthic taxa are represented in the river. Forty-three species of fish had been identified in the river (DOE, 1995c), and since then the brown bullhead has been collected, bringing the number to 44. Of these, 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. Small spring streams contain diverse biotic communities and are extremely productive, consisting of dense blooms of watercress and aquatic insects. Temporary wastewater ponds and ditches on the Hanford Site develop riparian communities and become quite attractive to migrating birds in autumn and spring.

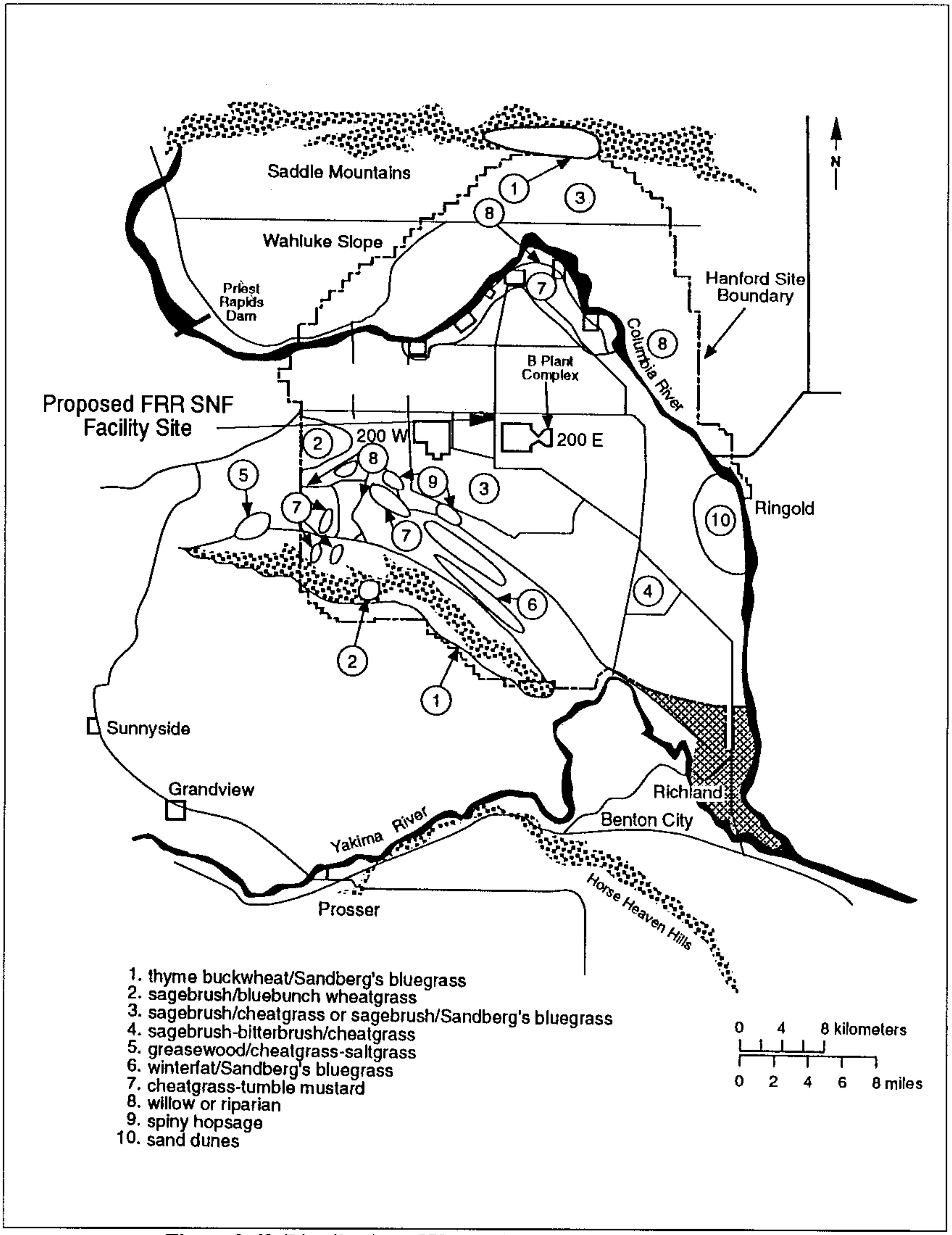


Figure 3-62 Distribution of Vegetation Types on the Hanford Site

Threatened, Endangered, and Candidate Plant and Animal Species: Threatened and endangered species, as listed by both the Federal Government and the State of Washington, are shown in Table 4.9-1 of Appendix A, Volume 1 of the Programmatic SNF&INEL Final EIS (DOE, 1995c). No plants or mammals on the Federal list are known to occur on the site. There are, however, several species of plants and animals that are under consideration for formal listing. Two species of plants are included in the State listing. Columbia milk-vetch, found on dry land benches along the river, is listed as threatened, and yellowcress, found on the wetted zone of the water's edge, is designated as endangered. The Federal Government lists the American peregrine falcon as endangered and the bald eagle as threatened. The State of Washington lists (in addition to the peregrine falcon and bald eagle) the white pelican and sandhill crane as endangered, and the ferruginous hawk as threatened. The peregrine falcon is a casual migrant to the Hanford Site and does not nest there. The bald eagle is a regular winter resident, foraging on dead salmon and waterfowl along the river, but not nesting on the Hanford Site. Ferruginous hawks have increasingly used power poles for nesting sites. Mammals considered endangered by the State are the pygmy rabbit, the Merriam shrew, the pallid bat, and the long-eared myotis.

Several small spring streams, wetlands, temporary water bodies, and national and State wildlife refuges are located on, or adjacent to, the Hanford Site.

3.3.3.6 Land Use

The Hanford Site encompasses 1,450 km^2 (560 mi^2), and includes several DOE operational areas. The major areas are:

- The entire Hanford Site, which has been designated a National Environmental Research Park;
- The 100 Areas, bordering on the right bank (south shore) of the Columbia River, which are the sites of the eight retired plutonium production reactors. The 100 Areas occupy about 11 km^2 (4.2 mi^2);
- The 200-West and 200-East Areas are located on a plateau about 8 and 11 km (5.0 and 6.8 mi), respectively, from the Columbia River. For some time, these areas have been dedicated to fuel reprocessing and waste processing management and disposal activities. The 200 Areas cover about 16 km^2 (6.2 mi^2);
- The 300 Area, located just north of the city of Richland, is the site of nuclear research and development and nuclear fuel fabrication. This area covers 1.5 km^2 (0.6 mi^2);
- The 400 Area covers about 0.6 km^2 (0.25mi^2) and is about 8 km (5 mi) north of the 300 Area and is the site of the Fast Flux Test Facility used in the testing of breeder reactor systems. Also included in this area is the Fuels and Material Examination Facility;
- The 600 Area includes all of the Hanford Site not occupied by the 100, 200, 300, or 400 Areas. Land uses within the 600 Area include:
  - The Arid Lands Ecology Reserve, a 310 km^2 (120 mi^2) tract set aside for ecological studies,
  - 4 km^2 (1.5 mi^2) leased by the State of Washington, part of which is used for low-level radioactive waste disposal,

- 4.4 km^2 (1.7 mi^2) for Washington Public Power Supply System nuclear power plants,
- 2.6 km^2 (1.0 mi^2) transferred to the State of Washington as a potential site for the disposal of nonradioactive hazardous wastes,
- About 130 km^2 (50 mi^2) under revocable use permit to U.S. Fish and Wildlife Refuge,
- 225 km^2 (87 mi^2) under revocable use permit to the Washington State Department of Game for recreational game management,
- Support facilities for the controlled access areas, and
- Laser Intercrometer Gravitational Wave Observatory.

Surrounding the Hanford Site, 660 km^2 (255 mi^2) have been designated for Arid Lands Ecology Reserve, U.S. Fish and Wildlife Refuge, and Washington State Department of Game (DOE, 1986a).

Land use in other areas includes urban and industrial development, irrigated and dry-land farming, and grazing. In 1985, wheat represented the largest single crop in terms of area planted in Benton and Franklin counties with 116,145 ha (287,000 acres). Corn, alfalfa, hay, and barley 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 (Richland, Pasco, and Kennewick), 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 ha was $1,640 ($664 per irrigated acre). The largest percentages 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, dry beans, asparagus, and pea seed.

3.3.3.7 Noise

Studies at the Hanford Site dealing with the propagation of noise have dealt primarily with occupational noise at work sites. Environmental noise levels have not been extensively evaluated because of the remoteness of most the Hanford activities, and isolation from receptors that are covered by Federal or State statutes. This discussion will focus on what little environmental noise data is available. The majority of available information consists of model predictions, which in many cases have not been verified, since these predictions indicate that the potential to violate Federal or State standards is remote or unrealistic.

The Noise Control Act of 1972 and its subsequent amendments (Quiet Communities Act of 1978, 42 USC 4901-4918, 40 CFR 201-211) directs the regulation of environmental noise to the State. The State of Washington has adopted RCW 70.107, which authorizes the Department of Ecology to implement rules consistent with Federal noise control legislation. RCW 70.107 and the implementing regulations embodied in WAC 173-60 through 173-70 define the regulation of environmental noise levels. Maximum noise levels are defined for the zoning of the area for the environmental designation for noise abatement. The Hanford Site is classified as a Class C environmental designation for noise abatement area on the basis of industrial activities. Unoccupied areas are also classified as Class C areas by default, because they are neither Class A (residential) nor Class B (commercial). Maximum noise levels are established based on the environmental designation for noise abatement classification of the receiving area and the source area (DOE, 1995c).

3.3.3.8 Transportation

The Tri-Cities serve as a regional transportation and distribution center with major air, land, and river connections. The Tri-Cities area has direct rail service, provided by Burlington Northern and Union Pacific, which connects the area to more than 35 States (Figure 3-63). Union Pacific operates the largest fleet of refrigerated railcars in the United States, and is essential to food processors that ship frozen food from this area. Passenger rail service is provided by Amtrak, which has a station in Pasco.

The Hanford Site infrequently uses docking facilities at the ports of Benton on the Columbia River. No barge accidents were reported in 1988 (DOE, 1995c).

Daily air passenger and freight services connect the area with most major cities through the Tri-Cities Airport in Pasco. The airport is currently served by two commuter-regional and two national airlines. The main runway is 2,350 m (7,755 ft) in length, and can accommodate landings and takeoffs by medium-range commercial aircraft, such as the Boeing 727-200 and Douglas DC-9. Two additional airports, located in Richland and Kennewick, are limited to serving private aircraft.

The Tri-Cities are linked to the region by five major roads. Route 395 joins the area with Spokane to the northeast. Both route 395 and route 240, which crosses through the Hanford Site, connect with Interstate 90 to the north. Route 12 links the region with Yakima to the northwest, Lewiston, ID to the east, and Walla Walla to the southeast. Finally, the area is linked to Interstate 84 to the south, via Interstate 82 and Route 14. Routes 240 and 24 traverse the Hanford Site and are maintained by the State of Washington. Other roads within the Hanford Site are maintained by DOE.

3.3.3.9 Socioeconomics

The level of operations at the Hanford Site plays a dominant role in the socioeconomics of the Tri-Cities and other parts of Benton and Franklin counties. The agricultural community also has a significant effect on the local economy. Any major changes in the Hanford Site operations will affect the Tri-Cities and other areas of Benton and Franklin counties.

Three major sectors have been the principal driving forces of the economy in the Tri-Cities since the early 1970's: (1) DOE and its contractors operating the Hanford Site, (2) Washington Public Power Supply System in its construction and operation of nuclear power plants, and (3) the agricultural community, including a substantial food processing component. Most of the goods and services produced by these sectors are exported outside 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. Three other components can be identified a contributors to the economic base of the Tri-Cities economy. These include other employers, such as: (1) Siemens Nuclear Power Corporation in North Richland, (2) Sandvik Special Metals in Kennewick, and (3) Boise-Cascade's Wallula corrugated paper mill, tourism, and Government transfer payments in the form of pension benefits.

The Hanford Site dominates the local employment picture with more than one-quarter of the total nonagricultural jobs in Benton and Franklin counties (15,552 out of 64,300), so the Hanford Site payroll impacts the Tri-Cities and State economy. In 1991, the Hanford Site employment accounted directly for

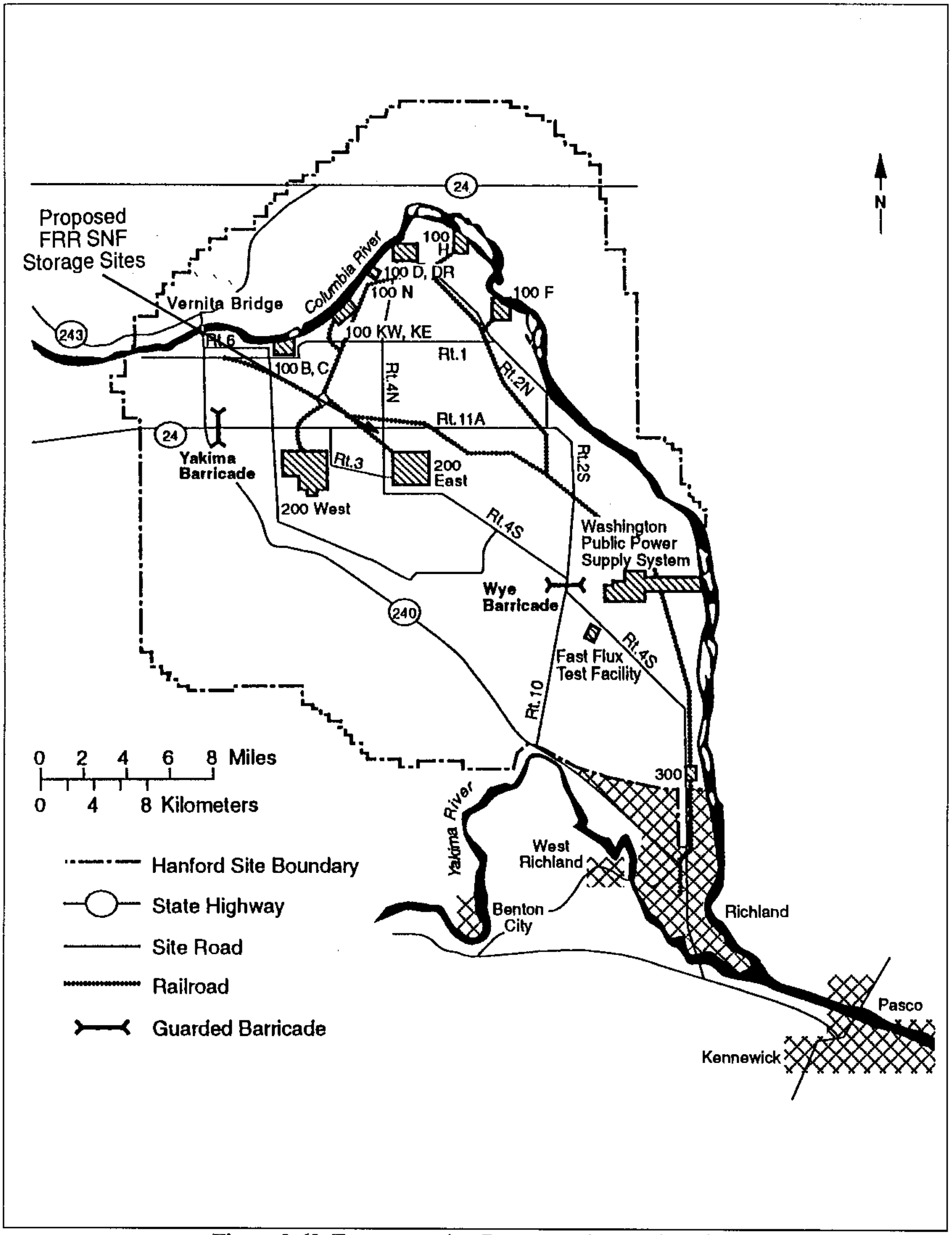


Figure 3-63 Transportation Routes on the Hanford Site

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

The Washington Public Power Supply System continues to be a major employer in Richland, with more than 1,700 workers and an approximate $71.6 million in payroll during the year. In 1990, agriculture was responsible for nearly 12,900 jobs, nearly 17 percent of total employment. This includes about 2,200 farm proprietors accounting for $121 million in crop and livestock production which provides 7,600 wage and salary jobs, and "agri-business" (farm and ranch supporting activities such as application of fertilizers, sales of farm supplies and equipment, etc.) accounting for 900 jobs. Employment in the food processing sector included 20 food processors producing potato products, canned fruits and vegetables, wine, and animal feed.

Other major employers include about 3,500 workers in Benton and Franklin counties. Tourism has increased in the area, and overall tourism expenditures in the Tri-Cities were roughly $77.5 million in 1990. In 1990, 15,903 people over the age of 65 resided in Benton and Franklin counties. 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. 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 of Appendix A, Volume 1 of the Programmatic SNF&INEL Final EIS (DOE, 1995c). The estimated income of this component of the basic sector is roughly equivalent to the entire agricultural sector.

Estimates by the U.S. Bureau of the Census for 1990 place Benton and Franklin counties' population totals at 112,560 and 37,473, respectively. Within each county, the 1990 estimates distribute the Tri-Cities population as follows: Richland, 32,315; Kennewick, 42,159; and Pasco, 20,337. The 1990 estimates of racial categories by the Bureau of the Census indicate that in Benton and Franklin counties, Asians represent a lower proportion, and individuals of Hispanic origin represent a higher proportion of the racial distribution than those in the State of Washington. Fifty-six percent of the population of Benton and Franklin counties is under the age of 35, compared to 54 percent for the State of Washington, and the 65-yr-old and older group constitutes 10 percent of the population, compared to 12 percent for the State.

Social and economic impacts of the Hanford Site operations are concentrated in Benton County, Franklin County, and the Tri-Cities area made up of Pasco, Richland, and Kennewick. The region of influence for the Hanford Site is represented by the 80 km (50 mi) radius around the site. Figure 3-64 represents the general ethnic characteristics of the population within the 80 km (50 mi) radius. Low income data for the region of influence is shown in Figure 3-65. A low-income household is one with an income of 80 percent or lower than the median income of the county. Approximately 42 percent of the households in the region of influence are low income families.

In 1990, nearly 92 percent of all housing (38,781 total units) in the Tri-Cities was occupied. Single-unit housing, which represents nearly 58 percent of the total units, has a 96 percent occupancy rate. Multiple-unit housing has an occupancy rate of nearly 91 percent. Pasco has the lowest occupancy rate, 89 percent in all categories of housing, followed by Kennewick (93 percent) and Richland (94 percent). Representing nine percent of the housing unit types, mobile homes have an 81 percent occupancy rate.

Primary and secondary education are served by the Richland, Kennewick, Pasco, and Kiona-Benton school districts. In 1992, spring enrollment for all districts was approximately 24,876 students. Post-secondary education in the Tri-Cities area is provided by a junior college, Columbia Basin College, and the Tri-Cities branch campus of Washington State University.

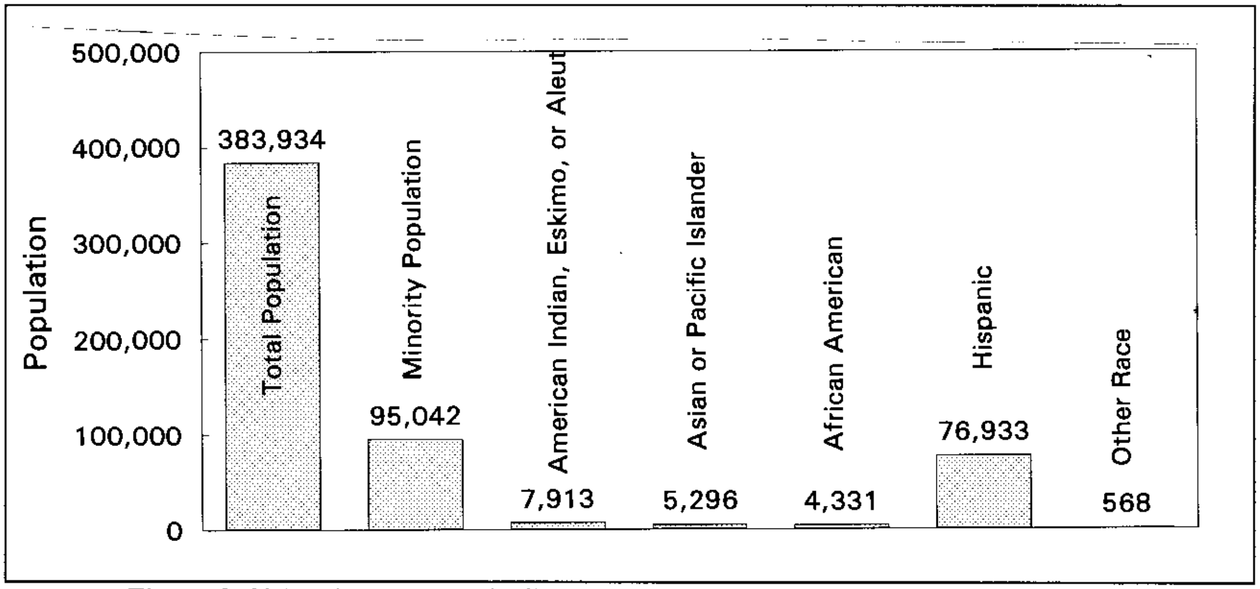


Figure 3-64 Racial and Ethnic Composition of the Minority Population Residing within 80 km (50 mi) of the Hanford Site

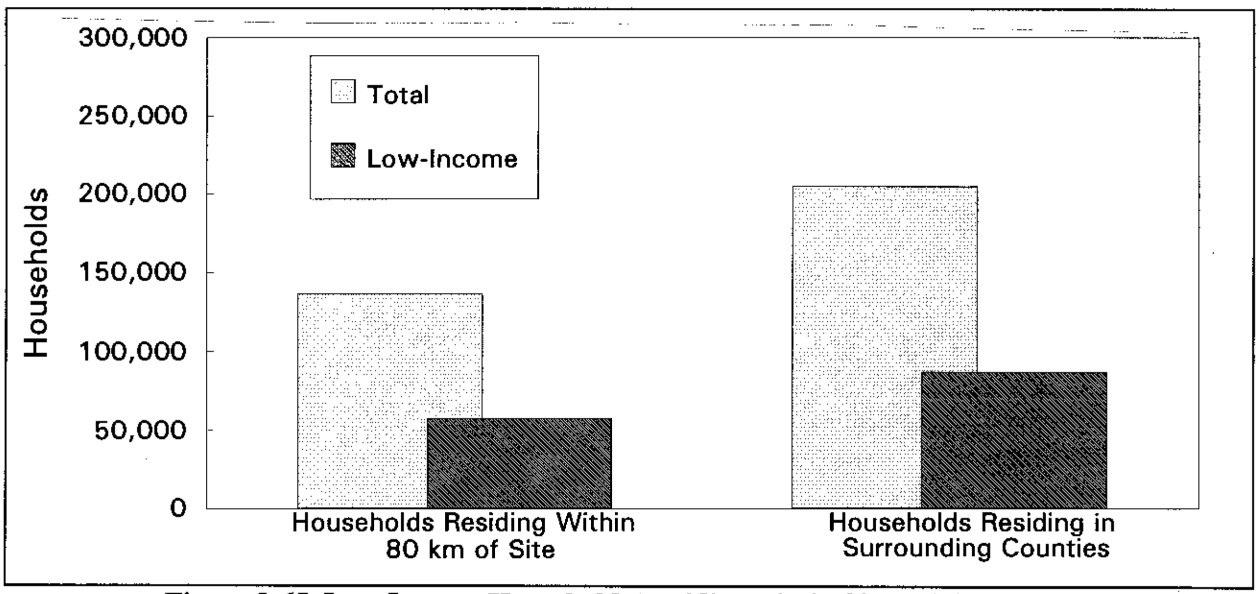


Figure 3-65 Low-Income Households Residing within 80 km (50 mi) of the Hanford Site

The Tri-Cities have three major hospitals and four minor emergency centers. Kadlec Medical Center, located in Richland, has 136 beds and functions at an average of 39.5 percent capacity. Kadlec Medical Center's 5,754 annual admissions represent more than 42 percent of the Tri-Cities market. Kennewick General Hospital maintains an average 45.5 percent occupancy rate of its 71 beds with 3,619 admissions. Our Lady of Lourdes Hospital, located in Pasco, reports an average occupancy rate of 36.5 percent.

Police and fire protection in Benton and Franklin counties is provided by Benton and Franklin county sheriff's departments, local municipal police departments, and the Washington State Patrol Division, headquartered in Kennewick. The Hanford Fire Patrol is composed of 126 trained firefighters.

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 43 billion liters (11.4 billion gal) used in 1991. Electricity is provided by the Benton County Public Utility District, Benton Rural Electric Association, Franklin County Public Utility District, and city of Richland Energy Services Department. In the Pacific Northwest, hydropower, and to a lesser extent, coal and nuclear power, constitute the region's electrical generation system. Throughout the 1980's, the Northwest had more electric power than it required, and was operating at a surplus. This surplus has been exhausted, and there is only approximately enough power supplied by the existing system to meet the current electricity needs.

3.3.3.10 Historical, Archaeological, and Cultural Resources

The Hanford Site is rich in cultural resources, and contains numerous well-preserved archaeological sites representing both prehistoric and historical periods. The area is considered a homeland by many Native Americans.

More than 10,000 years of prehistoric human activity in the Middle Columbia River region have left extensive archaeological deposits along river shores (DOE, 1995c) and well-watered areas inland from the river (DOE, 1995c; Greene, 1975;). Graves are common in various settings, and spirit quest monuments are found on high, rocky summits (Rice, 1968). By virtue of their inclusion in the Hanford Site, from which the public is restricted, archaeological deposits found in the Hanford reach of the Columbia River, on adjacent plateaus, and mountains have been spared some of the disturbances that have befallen other sites.

There are currently 248 prehistoric archaeological sites recorded in the files of the Washington State Office of Archaeology and Historic Preservation. Forty-seven of these sites are included on the National Register of Historic Places, two as single sites (45BN121, the Hanford Island Site; 45GR137, Paris Site), and the rest in seven archaeological districts. In addition, a nomination has been prepared for one cultural district (Gable Mountain/Gable Butte), and a renomination for two additional archaeological districts is pending (Wahluke, Coyote Rapids). Two other sites, 45BN90 and 45BN412, are considered eligible for the National Register. Archaeological sites include remains of numerous pithouse villages, various types of open campsites, and cemeteries along the riverbanks (DOE, 1995c), spirit quest monuments, hunting camps, game drive complexes and quarries in mountains and rocky bluffs, hunting/kill sites in lowland stabilized dunes, and small temporary camps near perennial sources of water located away from the river (Rice, 1968).

Native American people of various tribal affiliations have populated the Hanford reach of the Columbia River since prehistoric and early historic times. Wanapums and Yakama people of the Chamnapum band, and some of their descendants still live nearby, while 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 and inhabited the river's west bank (DOE, 1995c). Walla Walla and Umatilla people also made periodic visits to the area to fish. These groups retain traditional and secular ties to the region, with native plant and animal foods, some found on the Hanford Site, being used in ceremonies. Certain landmarks, especially Rattlesnake Mountain, Gable Mountain, Gable Butte, Goose Egg Hill and others along the river are considered sacred. The many cemeteries found along the river are considered sacred by these groups.

Two hundred-two historic archaeological sites and 11 other historic localities have been recorded in the published literature. 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 (DOE, 1995c). In addition to recorded sites, numerous data from additional unrecorded sites, including homesteads, corrals, and dumps, have been recorded by the Hanford Cultural Resources Laboratory since 1987. The 100-B Reactor has been listed on the National Register of Historic Places. Other Manhattan Project facilities remain to be evaluated.

3.3.4 Description of the Affected Environment at the Oak Ridge Reservation

The Oak Ridge Reservation is a key DOE site hosting three separate facilities with missions including basic and applied research and development; storage of special nuclear materials; weapons dismantlement, storage, and evaluation; and environmental restoration and waste management. The site is operated for DOE by Martin Marietta Energy Systems. This section describes the potentially affected environment at the Oak Ridge Reservation.

3.3.4.1 Geology

The Oak Ridge Reservation lies within the western portion of the Valley and Ridge Province, near the boundary with the Cumberland Plateau, in the State of Tennessee (Figure 3-66). The Valley and Ridge Province is characterized by numerous linear ridges and valleys that trend approximately southwest-northeast. A generalized geologic map of the Oak Ridge Reservation is shown in Figure 3-67. The Oak Ridge Reservation is mostly underlain by the Rome Formation and Conasauga, Knox, and Chickamauga Groups, sedimentary rocks of Cambrian and Ordovician age. A detailed description of these formations is given in the Programmatic SNF&INEL Final EIS (DOE, 1995c).

Sinkholes, large springs, caves, and other karst features are common in the Knox Group, and those parts of the Oak Ridge Reservation underlain by limestones and dolomites are classified as karst terrains. Although no site-specific geologic characterization has been conducted at the West Bear Creek Valley site, it appears that the proposed site for the construction of a foreign research reactor spent nuclear fuel storage facility is located over the lower Conasauga Group strata not normally characterized by karst development.

The soils found in the Oak Ridge Reservation vicinity generally contain clay minerals, chert, and quartz sand (Hatcher et al., 1992). Soils on the Oak Ridge Reservation tend to retain moisture, and are typically 90 percent saturated below a depth of 3 m (10 ft) (Ketelle and Huff, 1984). Depth of soil profiles on the Oak Ridge Reservation vary from 15 cm (5.9 in) on slopes, to 18 m (60 ft) over dolomites in the Knox Group (Boyle et al., 1982).

3.3.4.2 Seismology and Volcanology

The Oak Ridge Reservation is located in a region of moderate seismic activity, having an average of one to two earthquakes per year, with seismic activity occurring in bursts followed by long periods of inactivity. From 1811 to 1975, only five major earthquakes or earthquake series have affected the Oak Ridge Reservation area. No deformation of recent surface deposits has been detected, and seismic shocks from the surrounding, more seismically active areas, are dissipated by distance from the epicenters (Boyle et al., 1982). During the 1811 to 1975 period, none of the earthquakes were of a magnitude that caused severe damage to buildings or structures.

The underlying structure of the Oak Ridge Reservation is complex due to the extensive faulting and deformation characteristic of the region. There are three regional thrust faults in the Oak Ridge Reservation area, the Kingston, Whiteoak Mountain, and Copper Creek Faults. All three strike to the

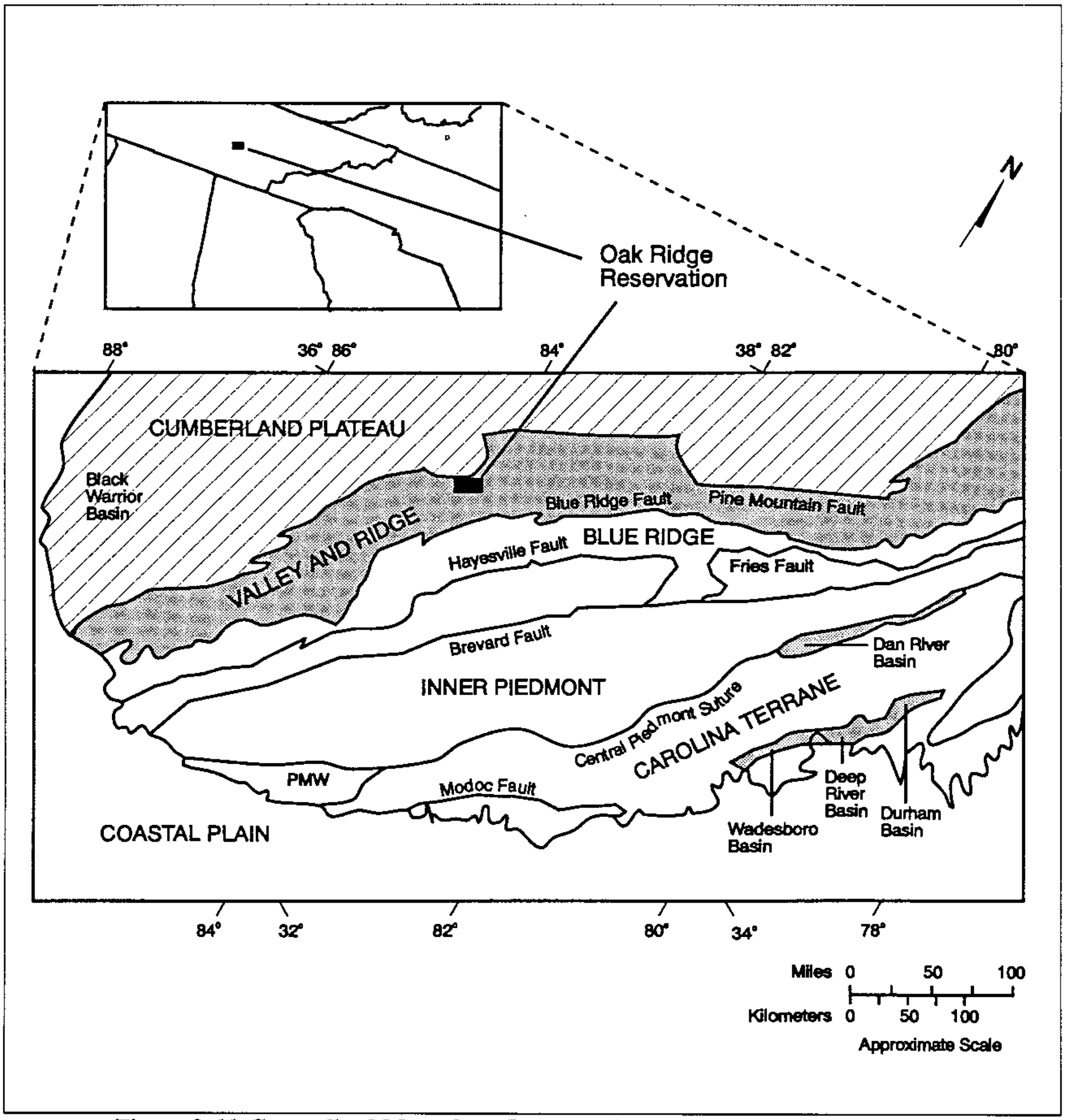


Figure 3-66 Generalized Map of the Southern Appalachian Geologic Provinces Showing the Location of the Oak Ridge Reservation

northeast and dip to the southeast. The most recent movement on the faults was during the Late Pennsylvanian/Early Permian periods (280 to 290 million years ago), and consequently, the faults are not considered to be capable at present (Butz, 1984).

There is no evidence that there has been significant volcanic activity in the vicinity of the Oak Ridge Reservation for more than 1 million years (DOE, 1995c). Three studies conducted specifically for the Oak Ridge Reservation have been summarized (Beavers et al., 1982).

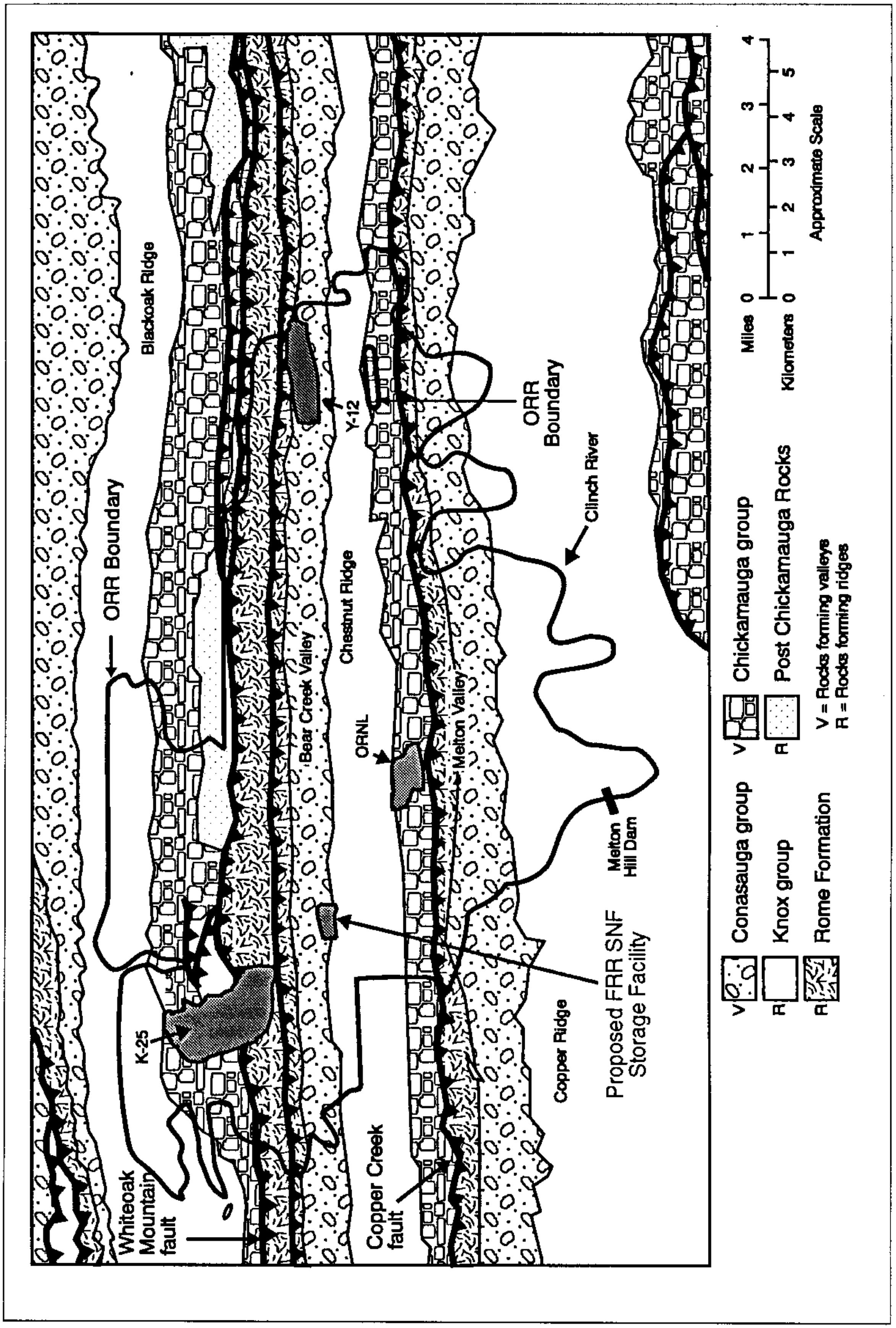


Figure 3-67 Geologic Map of the Oak Ridge Reservation

3.3.4.3 Hydrology

3.3.4.3.1 Surface Water

The hydrologic system on the Oak Ridge Reservation is controlled by the Clinch River (MMES, 1993a). Its drainage area is about 11,422 km^2 (4,410 mi^2) (Boyle et al., 1982). All water that drains from the Oak Ridge Reservation enters the Clinch River, and subsequently the Tennessee River. Flow in the Clinch-Tennessee river system is regulated by multi-purpose dams of the Tennessee Valley Authority.

The Oak Ridge Reservation is drained by a network of tributaries of the Clinch River. A Statewide stream classification system based on water quality, water use, and resident aquatic biota designates most streams on the Oak Ridge Reservation for fish and aquatic life, irrigation and livestock watering (MMES, 1993b). The drainage pattern on the Oak Ridge Reservation is a weakly developed "trellis" pattern (Lee and Ketelle, 1987), and the surface drainage patterns are also affected by karst topography.

Heavy precipitation in the area causes localized flooding, primarily in the city of the Oak Ridge (MMES, 1993a) and along the Clinch River. Figure 3-68 shows approximate 500-yr floodplains. A number of wetlands occur on the Oak Ridge Reservation (MMES, 1993b), including characteristic forested wetlands along creeks, wet meadows and marshes associated with streams and seeps, and emergent communities in shallow embayments and ponds.

Surface Water Quality: Water quality in the Clinch River is affected by the Oak Ridge Reservation activities, by contaminants introduced upstream from the Oak Ridge Reservation, and by flow regulation at the Tennessee Valley Authority dams. Stream impoundment has resulted in a rise in water temperatures, sediment retention, and contaminant adsorption.

The Clinch River supplies most of the water to the Oak Ridge Reservation, the city of Oak Ridge, and other cities along the river (MMES, 1993a). Major water uses in the Oak Ridge area include withdrawals for industrial and public water supplies, commercial and recreational navigation, and other recreational activities such as fishing, boating and swimming. Water for the city of Oak Ridge is withdrawn upstream from any of the Oak Ridge Reservation discharge points. Five public water supplies, including the cities of Kingston and Harriman, TN, are located downstream of the Oak Ridge Reservation (MMES 1993a). These are located 4 km (2.5 mi) above and 34 km (21 mi) below the mouth of Poplar Creek, respectively.

3.3.4.3.2 Groundwater

Groundwater beneath the Oak Ridge Reservation is heavily influenced by the site geologic structure (Solomon et al., 1992). Geologic units of the Oak Ridge Reservation are assigned to two broad hydrologic groups, Knox Aquifer and the Oak Ridge Reservation aquitards. These aquitards may store fairly large volumes of water, but they transmit only limited amounts.

The Knox Aquifer is the only true aquifer of the Oak Ridge Reservation, and is the primary source of sustained natural flow in perennial streams (Solomon et al., 1992). Flow volumes and potential groundwater flow path length in the Knox Aquifer are greater than in the aquitard. No spent nuclear fuel management facilities would be sited in areas overlying the Knox aquifer.

Recently published reports such as "Status Report; A Hydrologic Framework for the Oak Ridge Reservation", and "Status Report on the Geology of the Oak Ridge Reservation" have all used the term "aquitard" to describe the Pumpkin Valley Shale and a number of the other geologic units beneath the Oak Ridge Reservation. No site-specific data are available to determine at what depth Pumpkin Valley Shale is

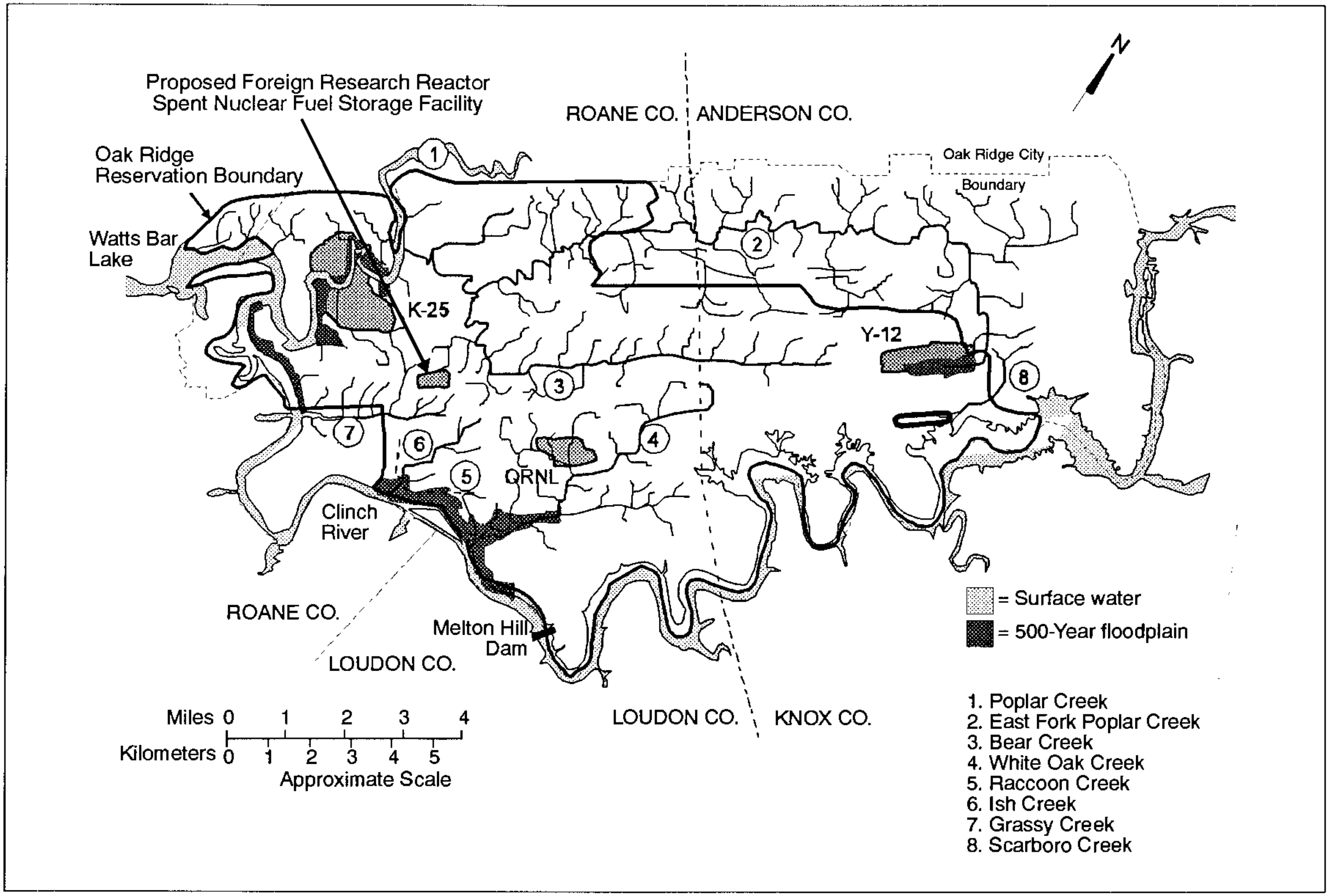


Figure 3-68 Locations of the Clinch River and Tributaries on the Oak Ridge Reservation

encountered at the West Bear Creek Valley, the proposed site for the construction of foreign research reactor spent nuclear fuel storage facilities. It would be logical, however, to think that at depths of 18 m (60 ft) or less on the site, the water-bearing unit most likely to be encountered would be an aquitard (DOE, 1995c).

Groundwater Quality: Background groundwater quality at the Oak Ridge Reservation is generally good in the surficial aquifer zones, and poor in the bedrock aquitards at depths greater than 305 m (1,000 ft) (DOE, 1993d). Groundwater has been contaminated downgradient from waste management facilities and other industrial sources, and discharge of contaminated groundwater has introduced contaminants to streams. Principal groundwater contaminants that exceed applicable standards at the Y-12 Plant include volatile organics, nitrates, heavy metals, and radioactivity (MMES, 1993b). There is one known instance of offsite migration of contaminants from the Oak Ridge Reservation. In 1994, DOE announced that elevated levels of four industrial solvents (carbon tetrachloride, chloroform, tetrachloroethylene, and trichloroethylene) had been detected in the Knox Aquifer monitoring wells east of the Y-12 Plant (Bowdle, 1994). The same solvents are found in groundwater monitoring wells within the Y-12 Plant.

There are no sole-source aquifers beneath the Oak Ridge Reservation (DOE, 1993d). Water rights are not an issue in the region. All groundwater at the Oak Ridge Reservation is classified as Class II (current and potential sources of drinking water and those waters having other beneficial uses).

Although surface water sources provide the main portion of potable water supplies in the area, groundwater does provide for some domestic, municipal, farm, irrigation, and industrial use (MMES, 1993b). Single-family wells are common in areas not served by public water supplies (MMES, 1992). Due to the abundance of surface water and its proximity to the points of use, almost no groundwater is used at the Oak Ridge Reservation (DOE, 1993d).

3.3.4.4 Meteorology

The Oak Ridge Reservation is located within the Great Valley of Tennessee, in which Cumberland Plateau borders to the northwest and the Great Smoky Mountains lie to the southeast. The climate at the Oak Ridge Reservation is influenced by these terrain features.

Wind: The wind direction above the ridgetops and within the valleys tends to follow the orientation of the valleys. The prevailing wind direction is from the southwest, with a secondary maximum from the northeast during the winter, spring, and summer months. This situation is reversed in the fall. Damaging winds are uncommon in the region. Peak gusts recorded in the Great Valley are generally in the 27 to 31 m per sec (60 to 70 mph) range for the months of January through July, in the 22 to 27 m per sec (50 to 60 mph) range for August, September, and December, and in the 16 to 20 m per sec (36 to 45 mph) range in October and November.

Temperature and Humidity: The average daily temperature at the Oak Ridge National Weather Service Station was 14.2°C (57.6°F) for the period of record 1961-1990. The average daily temperature varied from a low of 2.6°C (36.7°F) in January to a high of 24.8°C (76.6°F) in July. The mean relative humidity was 86 percent, with the mean monthly maximum of 92 percent occurring in July and August, and the mean monthly minimum of 80 percent occurring during February and March.

Precipitation: Winter is the wettest of the seasons in the Oak Ridge Reservation area, March and December are the wettest months, and October is the driest. The maximum monthly precipitation was 48.9 cm (19.3 in) in July, while maximum rainfall in a 24-hr period observed at the Oak Ridge National Weather Service was 19 cm (7.5 in), recorded in August 1960 (DOE, 1995c). The annual average precipitation reported by the National Weather Service for the Oak Ridge was 137.2 cm (54 in).

On average, about 51 thunderstorm days per year are recorded at the Oak Ridge National Weather Service station. The Great Valley of Tennessee is infrequently subject to tornadoes. The western half of the State has experienced three times as many tornadoes as the eastern half, where the Oak Ridge Reservation is located. The Oak Ridge Reservation did experience a tornado from a severe thunderstorm on February 21, 1993. The tornado path passed the Y-12 Plant in an east-northeast direction for approximately 21 km (13 mi), ending just north of Knoxville. The wind speeds associated with this tornado ranged from 18 m per sec (40 mph) to nearly 58 m per sec (130 mph), depending on the location along the path (DOE, 1993c). Hurricanes are rarely sustained once they reach as far inland as the Great Valley, due to the rapid loss of energy.

Atmospheric Dispersion: The transport and dispersion of airborne material are direct functions of air movement. The atmospheric conditions are unstable (Stability Classes A through C) approximately 5 percent of the time, neutral (Class D) approximately 43 percent of the time, and stable (Classes E through G) approximately 52 percent of the time at the 10 m (33 ft) level.

Air Quality: A summary of the Oak Ridge Reservation airborne radionuclide emissions for 1992 is presented in Table 4.7-1 of Appendix F, Volume 1 of the Programmatic SNF&INEL Final EIS (DOE, 1995c). The maximum effective dose equivalent at the Oak Ridge Reservation boundary is 3.3 mrem as calculated by the GENII code. This is 33 percent of the National Emissions Standards for Hazardous Air Pollutants.

The Oak Ridge Reservation is located in Anderson and Roane Counties in the Eastern Tennessee-Southwestern Virginia Interstate Air Quality Control Region 207. As of 1993, the areas within this Air Quality Control Region were designated as attainment with respect to all National Ambient Air Quality Standards (40 CFR 81.343).

3.3.4.5 Ecology

Land for the Oak Ridge Reservation was primarily in agricultural use, including woodlots and woodlands used for pasture, at the time of acquisition by DOE's predecessor agencies. At least half of the Oak Ridge Reservation was forested. Most of the forest had been partially cut for timber, but not completely cleared on steep slopes. Natural plant communities have re-established themselves on most of the Oak Ridge Reservation, although many areas are maintained as pine plantations or nonforested areas (ORNL, 1988). Approximately 10 percent of the Oak Ridge Reservation has been developed since it was withdrawn from public access, and the remainder of the site has reverted to or been planted with natural vegetation (MMES, 1989).

The vegetation of the Oak Ridge Reservation has been categorized into seven plant communities (Parr and Pounds, 1987). The Oak Hickory forest is one of the most extensive plant communities on the Oak Ridge Reservation. Another abundant plant community is the Pine Hardwood forest and Pine plantations. A total of 899 species, subspecies, and varieties of plants have been identified on the Oak Ridge Reservation (Cunningham and Pounds, 1991). The Oak Ridge Reservation also provides habitat for a large number of animal species. Twenty-six species of amphibians, 33 species of reptiles, 169 species of birds, and 39 species of mammals have been recorded at the Oak Ridge Reservation (Parr and Evans, 1992).

Vegetative communities of the West Bear Creek site are typical of the Oak Ridge Reservation as a whole, composed of second-growth oak-hickory forest and mixed pine-hardwood forest. There are some loblolly pine plantations adjacent to the northern edge of the powerline right-of-way and between the right-of-way and Bear Creek Road (Rosensteel, 1994). Fauna of the site would also be similar to those expected throughout the Oak Ridge Reservation.

Wetlands on the Oak Ridge Reservation include emergent, scrub/shrub, and forested wetland located in embayments of the Melton Hill and Watts Bar Reservoirs that border the Oak Ridge Reservation, along all the major streams, including East Fork Poplar Creek, Poplar Creek, Bear Creek, and their tributaries, in old farm ponds, and around groundwater seeps. Originating on the lower slopes of Pine Ridge are several headwater tributary systems on Grassy Creek that flow from north to south across the West Bear Creek site. The stream valleys contain forested wetlands.

Aquatic habitats on or adjacent to the Oak Ridge Reservation range from small, free-flowing streams in undisturbed watersheds to larger streams with altered flow patterns due to dam construction. These aquatic habitats include tailwaters, impoundments, reservoir embayments, and large and small perennial streams.

A National Environmental Research Park Aquatic Reference Area is located along Grassy Creek and its tributaries, one of which runs through the eastern portion of the proposed spent nuclear fuel management site. Grassy Creek has a diverse assemblage of invertebrates and fish species for a stream its size. The Oak Ridge Reservation uses Grassy Creek as a reference area for studies of other streams affected by site development (Pounds et al., 1993).

Threatened, Endangered, and Candidate Plant and Animal Species: No animal species listed by the Federal Government as threatened or endangered are known to reside on the Oak Ridge Reservation (Kroodsma, 1987). The bald eagle is a winter visitor to Watts Bar Lake and Melton Hill Lake. None of the species listed in Table 4.9-1 of Appendix F, Volume 1 of the Programmatic SNF&INEL Final EIS have been recorded on the proposed West Bear Creek Valley site (DOE, 1995c). Table 4.9-1 of Appendix F, Volume 1 of the Programmatic SNF&INEL Final EIS lists Federally and State-listed threatened, endangered or other special-status species designated by the Endangered Species Act and/or the State's Nongame and Endangered Species and the Rare Plant Protection and Conservation laws (DOE, 1995c). Preferred habitat within the site indicates a greater potential for occurrence of the barn owl, black vulture, Cooper's hawk, red-shouldered hawk, and sharp-shinned hawk (DOE, 1995c). No intensive threatened and endangered species surveys have been completed for the site, but they are currently in progress for the entire the Oak Ridge Reservation (King et al., 1994).

3.3.4.6 Land Use

The Oak Ridge Reservation occupies an area of approximately 14,029 ha (34,667 acres) in eastern Tennessee, in a predominantly rural area about 40 km (25 mi) west of Knoxville. The Oak Ridge Reservation is within the jurisdictional boundaries of the city of Oak Ridge, and also lies within Roane and Anderson Counties (MMES, 1989).

Land use activities at the Oak Ridge Reservation have historically occurred within the boundaries of the three main plant sites (Y-12, the Oak Ridge National Laboratory, and K-25). The Oak Ridge Reservation has been used by research institutions, universities, and Government agencies as a site for the study of terrestrial ecology, aquatic ecology, forestry, and agriculture. Land uses bordering the Oak Ridge Reservation are primarily forest and agricultural. Residential and commercial are the only other significant uses of land in the vicinity, and occur along the northeast and northwest boundaries of the Oak

Ridge Reservation in the city of Oak Ridge. Figure 3-69 shows the land use in and around the Oak Ridge Reservation. The entire Oak Ridge Reservation has been placed under the forestry, agriculture, industry, and research zoning classification by the city of Oak Ridge. This classification does not bind DOE land use decisions on the site.

The region surrounding the Oak Ridge Reservation has numerous local, State, and national public recreation areas. Several lakes exist within the region surrounding the Oak Ridge Reservation, offering year-round recreational activities such as fishing and boating. The Oak Ridge Reservation is a controlled area, with public access limited to through traffic on Tennessee State Routes 95, 58, 62, 162, and 170 (MMES, 1991). There are no onsite areas that are subject to Native American Treaty rights or that contain any prime or unique farmland.

3.3.4.7 Noise

The major noise sources within the Oak Ridge Reservation occur primarily in developed operational areas and include various facilities, equipment, and machines. Major noise sources outside the operational areas consist primarily of vehicles and railroad operations. At the site boundary, noise from these sources would be barely distinguishable from background noise levels. The State of Tennessee has not established specific numerical environmental noise standards applicable to the Oak Ridge Reservation. The acoustic environment along the Oak Ridge Reservation site boundary is typical of a rural location, with the average soundlevel in the range of 35 to 50 decibels, A-weighted.

3.3.4.8 Transportation

The region of influence for the Oak Ridge Reservation includes site roads and regional roads up to approximately 24 km (15 mi) in Anderson, Blount, Knox, Loudon, and Roane counties. Primary roads on the Oak Ridge Reservation include Tennessee State Routes 95, 58, 62, 162, and 170 (Bethel Valley Road), and Bear Creek Road. Except for Bear Creek Road east of Route 95, all are public roads. The remaining roads on the Oak Ridge Reservation are private. Interstates 75 and 40, and Tennessee State Routes 162, 62, and 61 form a loop around the Oak Ridge Reservation (Figure 3-70).

Current baseline traffic (i.e., 1994) along segments providing access to the Oak Ridge Reservation is projected to contribute to differing service level conditions (TDOT, 1991). Tennessee State Route 61 would operate at level of service D between Interstate 75 at Norris and U.S. Route 25W at Clinton, and at level of service C between U.S. Route 25W at Clinton to Tennessee State Route 62 east of Oliver Springs. Tennessee State Routes 58 and 170 (providing access from the east), as well as Bear Creek Road, would operate between levels of service C and A. Tennessee State Routes 62 and 95 would operate at widely varying levels of service in the vicinity of the Oak Ridge Reservation. Tennessee State Route 62 would operate at a level of service E between Tennessee State Route 95 at Oak Ridge and Tennessee State Route 170 between Tennessee State Route 170 and Tennessee State Route 162. Tennessee State Route 95 would operate at a level of service E between Tennessee State Route 61 and Tennessee State Route 62 at Oak Ridge.

No public transportation service exists on the Oak Ridge Reservation. Other modes of transportation within the region of influence include railways and waterways. Railroad service in the region of influence is provided by CSX Transportation and the Norfolk Southern Corporation. Waterborne transportation is via the Clinch River. The Clinch River waterway has rarely been used for DOE business, and no designated port facilities exist for such purposes (U.S. Army Corps of Engineers, 1994).

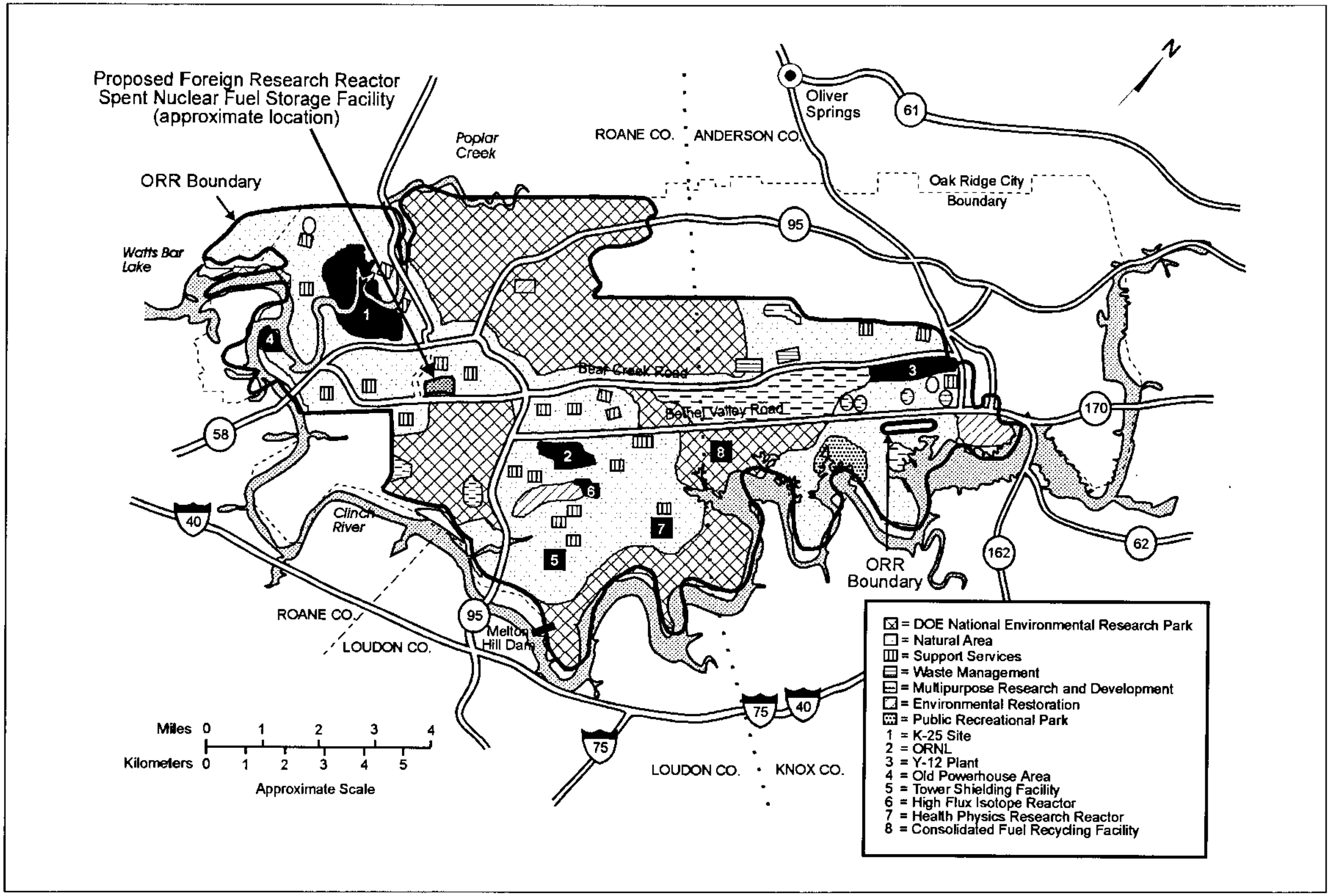


Figure 3-69 Generalized Land Use at the Oak Ridge Reservation

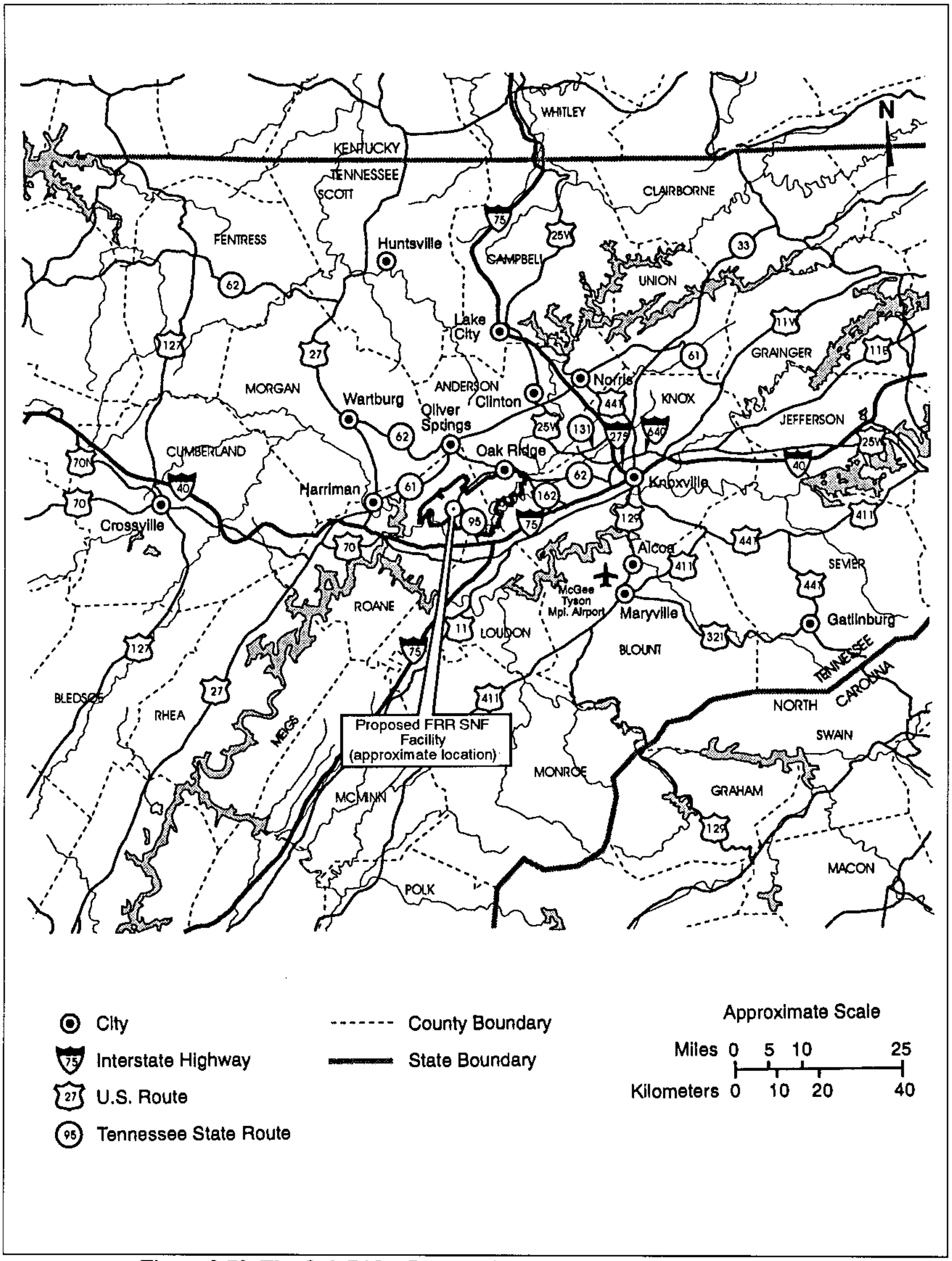


Figure 3-70 The Oak Ridge Reservation Regional Transportation Map

McGhee Tyson Airport in Knoxville, 64 km (40 mi) from the Oak Ridge Reservation, receives jet air passenger and cargo services from both national and international carriers. The closest air transportation facility to the Oak Ridge Reservation is Atomic Airport in Oliver Springs. Numerous other private airports are located throughout the region of influence (DOE, 1995c).

3.3.4.9 Socioeconomics

The region of influence includes the current residential distribution of DOE and contractor personnel employed by the Oak Ridge Reservation, the probable location of offsite contractor operations, and the probable location of labor and capital supporting indirect economic activity linked to the Oak Ridge Reservation. The region of influence includes the counties where 92 percent of DOE and contractor personnel employed by the Oak Ridge Reservation reside.

Regional economic linkage supporting production activity at the Oak Ridge Reservation occurs primarily with Anderson, Knox, Loudon, and Roane counties, where most of the offsite supporting contractors and labor and capital supporting indirect economic activity linked to the Oak Ridge Reservation are located. The total population of the region of influence is projected to be 489,230 persons in 1995 (DOE, 1995c), and is projected to grow at an annual average rate of less than 1 percent, reaching 538,820 persons in 2004. The labor force is also projected to grow at an annual average rate of less than 1 percent, growing to 360,000 persons in 2004. The total employment is projected to grow at an annual average rate of approximately 1 percent, growing from 292,700 jobs in 1995 to 338,070 jobs in 2004.

Figure 3-71 shows the racial and ethnic composition of minorities within 80 km (50 mi) of spent nuclear fuel management sites on the Oak Ridge Reservation. In comparison with the other four candidate sites, the Oak Ridge Reservation has the smallest percentage, about 6 percent, of minorities in the population residing around the site. African Americans make up approximately 76 percent of the minority population, while Hispanics and Asian Americans make up 8 to 9 percent of the minority population.

Figure 3-72 presents the low-income households residing within 80 km (50 mi) of the Oak Ridge Reservation. About 44 percent of the households are classified as having an income no larger than 80 percent of the median income for the county of residence. This percentage is typical of that for counties within 80 km (50 mi) of the spent nuclear fuel management sites.

The Oak Ridge Reservation fire protection services are provided by the fire departments on the reservation. The Oak Ridge Reservation fire departments have mutual aid agreements among themselves and with the city of Oak Ridge (DOE, 1995c). Twelve city, county, and State law enforcement agencies provide police protection in the region of influence. Law enforcement on the Oak Ridge Reservation is provided by the city of Oak Ridge Police Department. Security enforcement is provided by the prime management and operations contractor (MMES, 1989). Four county school districts (Anderson, Knox, Loudon, and Roane) and five city school districts (Clinton, Oak Ridge, Lenoir City, Kingston, and Harriman) provide public education services in the region of influence. In 1992, the nine school districts had an average daily membership of 75,825 students. Between 1980 and 1990, the number of housing units in the region of influence increased 14 percent from 181,299 to 206,234. In 1980 and 1990, the homeowner vacancy rates in the region of influence averaged 1.4 and 1.5 percent, respectively (DOE, 1995c).

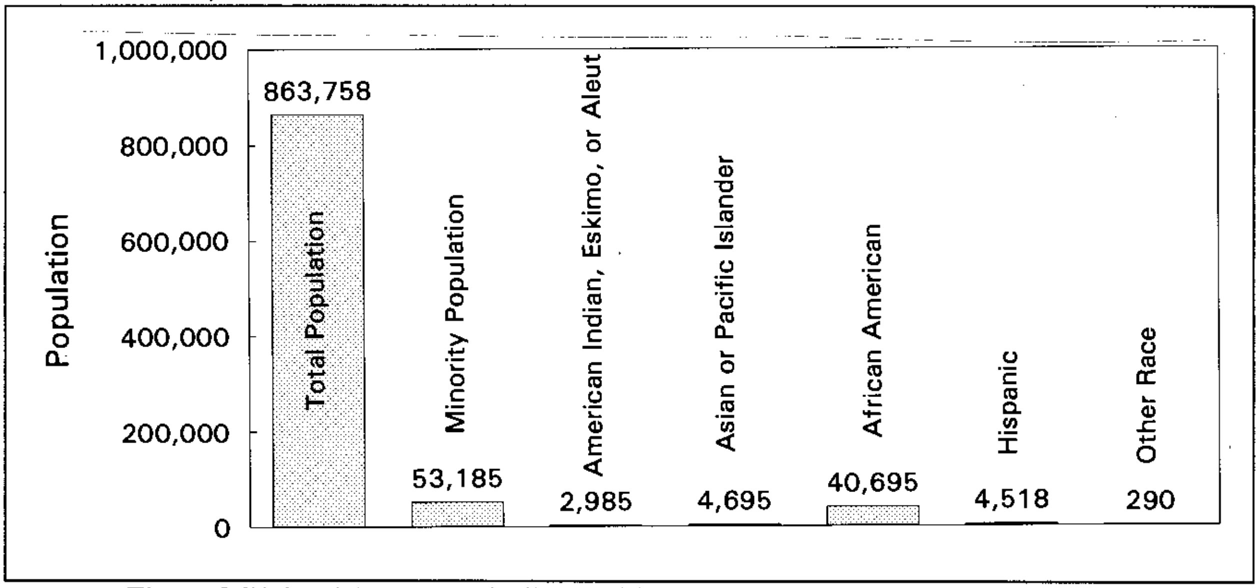


Figure 3-71 Racial and Ethnic Composition of the Minority Population Residing within 80 km (50 mi) of the Oak Ridge Reservation

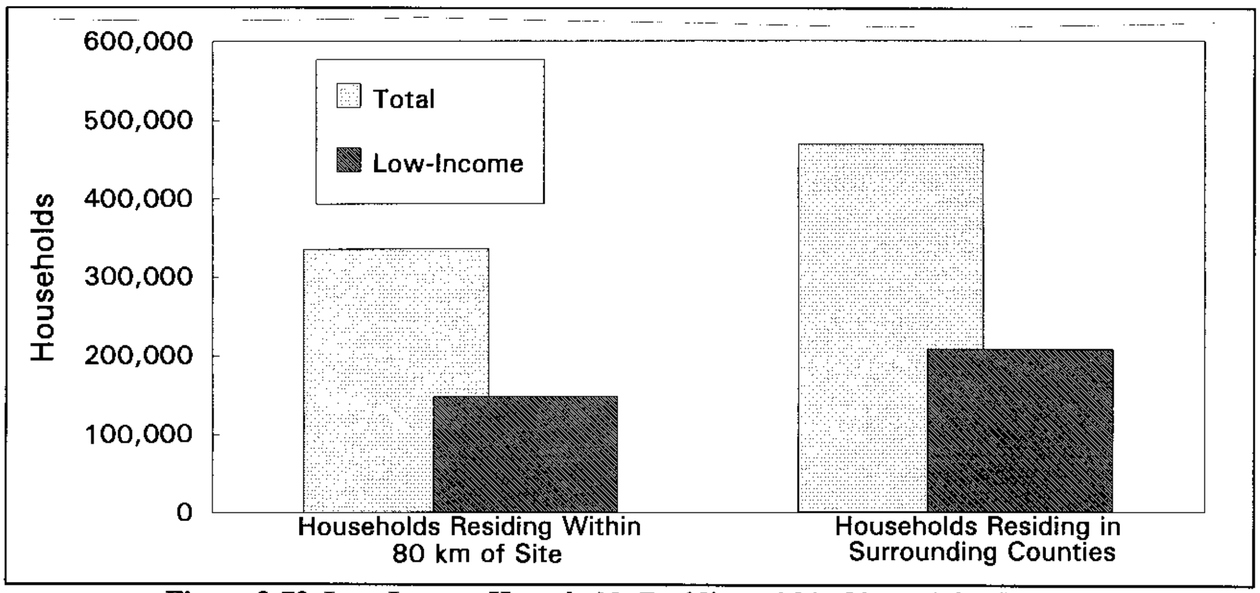


Figure 3-72 Low-Income Households Residing within 80 km (50 mi) of the Oak Ridge Reservation

3.3.4.10 Historical, Archaeological, and Cultural Resources

A limited survey conducted in 1975 did not identify any cultural resources on the Oak Ridge Reservation at the West Bear Creek Valley site (DOE, 1995c). No prehistoric or historic resources are expected to be located on the site. There are no known Native American resources on the proposed site.

3.3.5 Description of the Affected Environment at the Nevada Test Site

The Nevada Test Site is primarily used for the development and testing of nuclear weapons. This section describes the potentially affected environment of the site. The location of the site is shown in Figure 3-73.

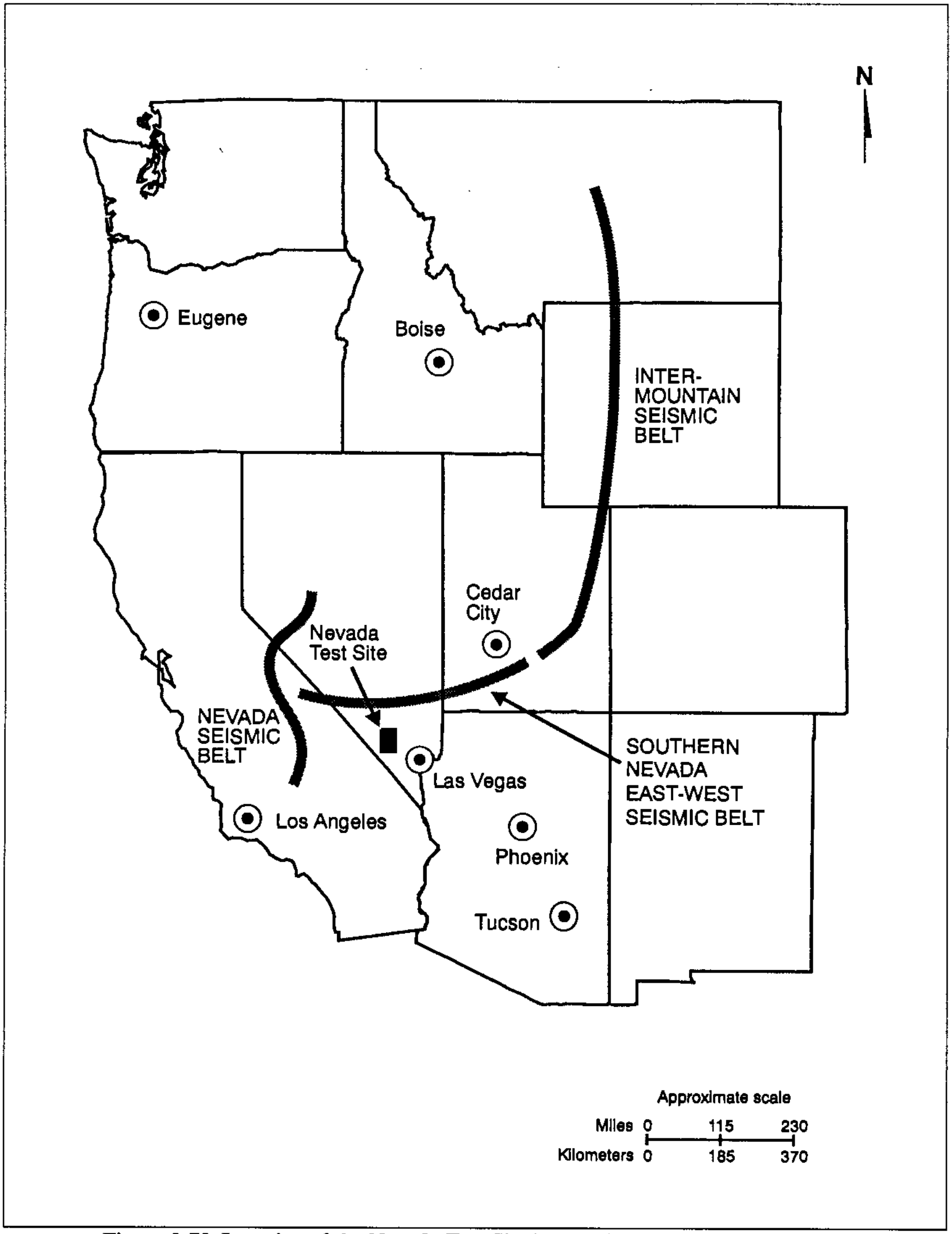


Figure 3-73 Location of the Nevada Test Site in Relation to the Nevada Seismic Belt, the Intermountain Seismic Belt, and the Southern Nevada East-West Seismic Belt

3.3.5.1 Geology

The Nevada Test Site is located east and north of the Walker Lane-Las Vegas Valley Shear Zone (Eckel, 1968). Walker Lane is a northwest-trending belt of right-lateral faults that disrupts the regional structural grain in the southwestern part of the Great Basin along the California-Nevada border. The Las Vegas Valley Shear Zone is a concealed zone of right-lateral faulting along the north side of the Las Vegas Valley (DOE, 1988a). The local geology of the Nevada Test Site is characterized by mountain ranges composed of Precambrian and Paleozoic sedimentary rocks and Tertiary volcanic tuffs and lavas that surround alluvium-filled, topographically closed valleys. A geologic map of the site is shown as Figure 3-74 (DOE, 1993b). The sedimentary rocks are complexly folded and faulted, and are composed mainly of carbonates (dolomite and limestone) in the upper and lower parts of the column and clastics (shale and sandstone) in the middle section. The volcanic rocks are predominantly tuffs that are high in silica.

Faulting generally occurs as thrust faults, normal faults, and strike-slip faults (DOE, 1992c). The faults are shown in Figure 3-75 (DOE, 1993b). Thrust faulting occurs as three major thrust faults, and normal faults exist in both ranges and valleys and generally strike northeast and northwest. The nearest strike-slip structure to the Nevada Test Site is the Walker Lane-Las Vegas Valley Shear Zone. Estimates of horizontal displacement along this shear zone range from 40 to 160 km (25 to 100 mi) (DOE, 1982). Recent displacements have occurred along several faults as a consequence of underground nuclear explosions. This disturbance is not attributable to naturally-occurring seismic activity. Fault displacements are thought to have occurred as a result of the added stress produced by the explosions, the vibrations produced by the explosions, or a combination of both (Eckel, 1968). Almost all of the natural fault movement in the Nevada Test Site area occurred several million years ago, except movement along Yucca Fault. Movement in the Yucca Fault is believed to have occurred in the past tens of thousands to 250,000 years (DOE, 1982; DOE, 1995c).

3.3.5.2 Seismology

The Nevada Test Site lies on the southern margin of the Southern Nevada East-West Seismic Belt (Figure 3-73), which is an area of relatively low historical seismicity. The regional seismicity is dominated by high-levels of seismic activity. Between 1978 and 1981, no earthquakes with magnitudes greater than 4.3 were recorded (DOE, 1986b). In 1992, a magnitude 5.6 earthquake was recorded near Little Skull Mountain at a depth of 12 km (7.5 mi). In 1993, a magnitude 3.5 earthquake was recorded southeast of the town of Mercury on the Nevada Test Site (DOE, 1995c).

The most probable source for seismic activity at the Nevada Test Site is the Cane Spring fault (Figure 3-75). Estimates of recurrence intervals for major earthquakes in the region are on the order of 25,000 years. For magnitudes of greater than or equal to 6, recurrence intervals are on the order of 2,500 years, and for magnitudes of greater than or equal to 5, recurrence intervals are on the order of 250 years (DOE, 1986b).

3.3.5.3 Hydrology

3.3.5.3.1 Surface Water

The drainage basins and the generalized directions of surface water flow near the Nevada Test Site are shown in Figure 3-76 (USAF et al., 1991). The boundary lines of the drainage basins occur principally along topographic divides (DOE, 1988a). Almost all streamflow in the Nevada Test Site area is

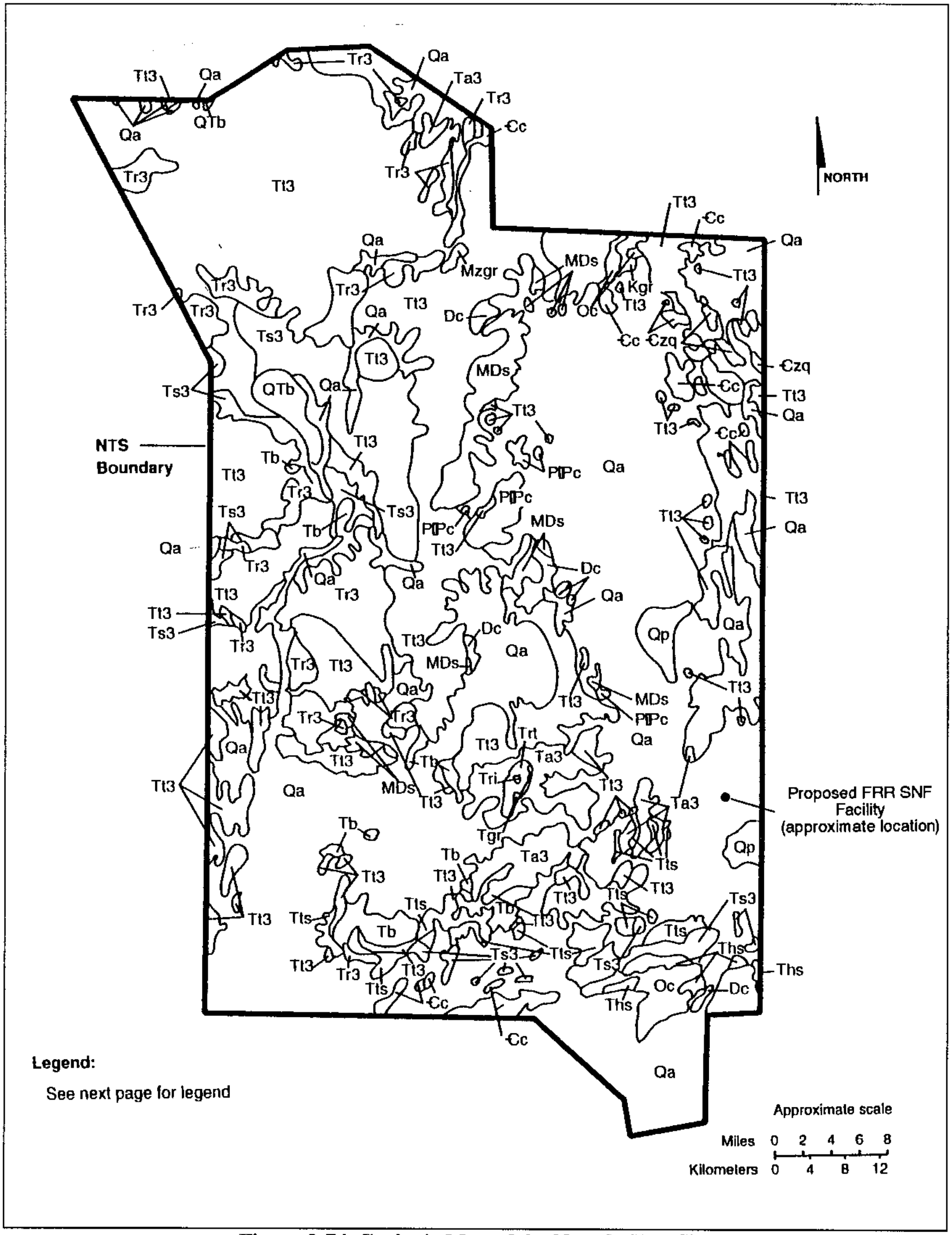


Figure 3-74 Geologic Map of the Nevada Test Site

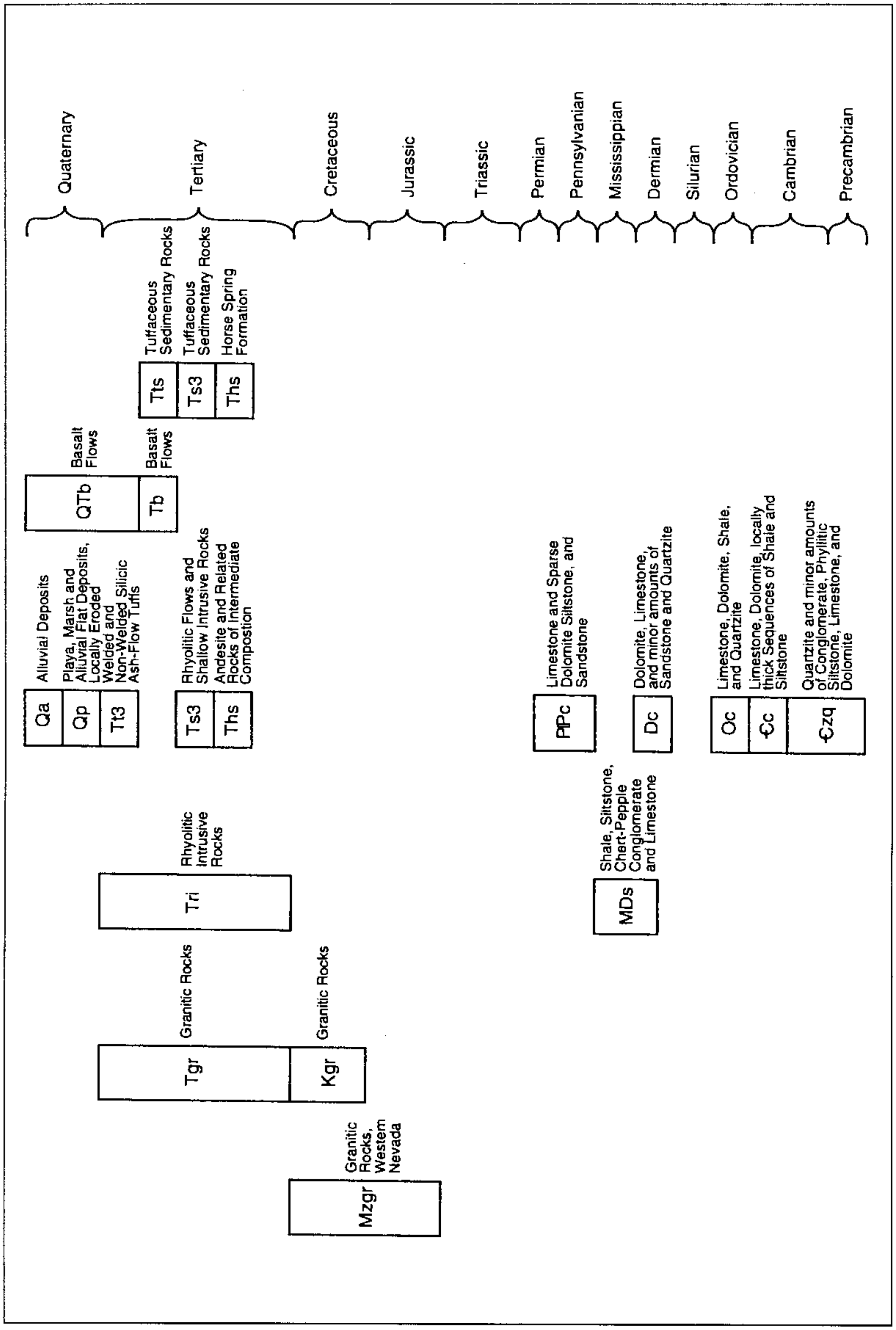


Figure 3-74 Geologic Map of the Nevada Test Site (Continued)

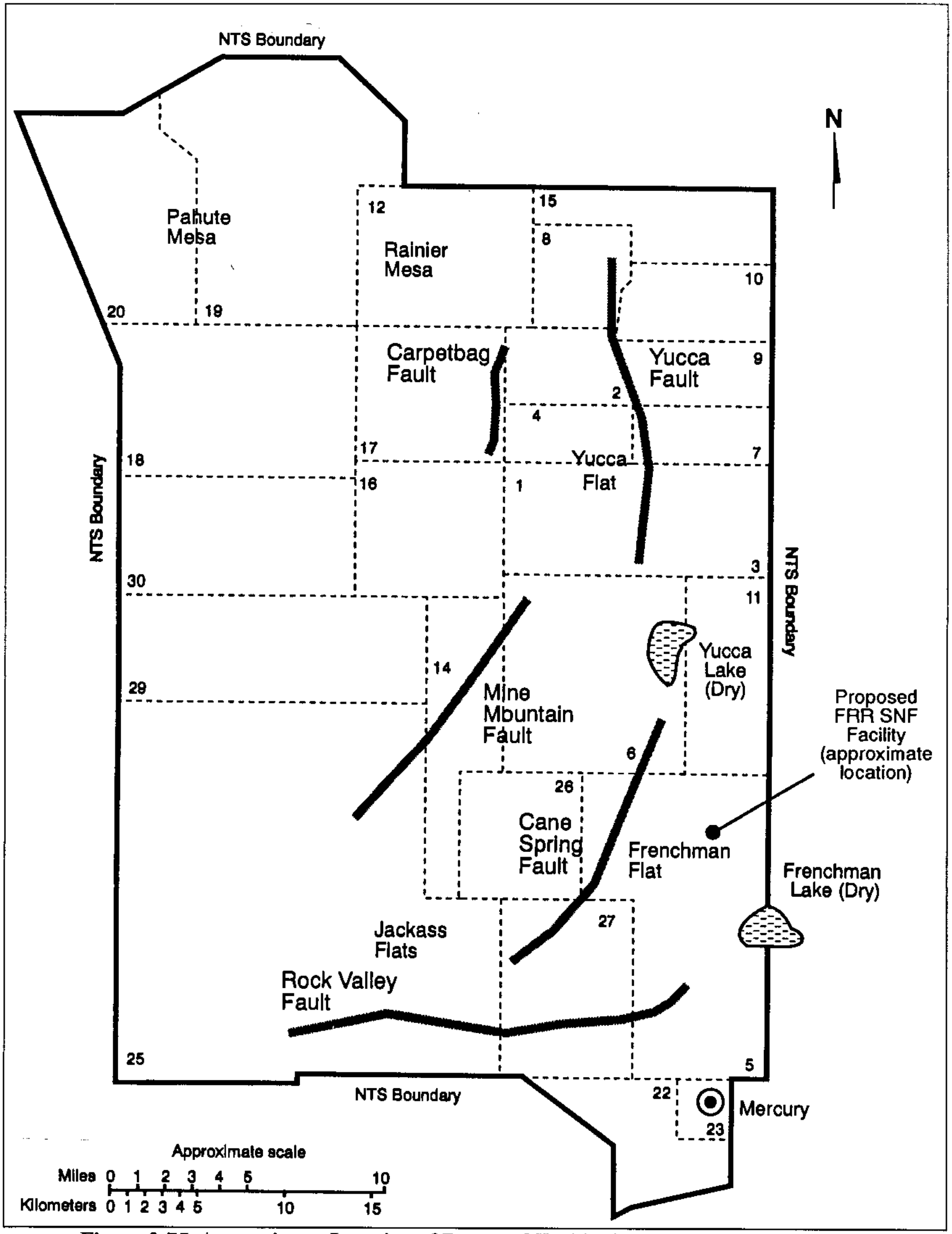


Figure 3-75 Approximate Location of Proposed Facility in Relation to Major Faults at the Nevada Test Site

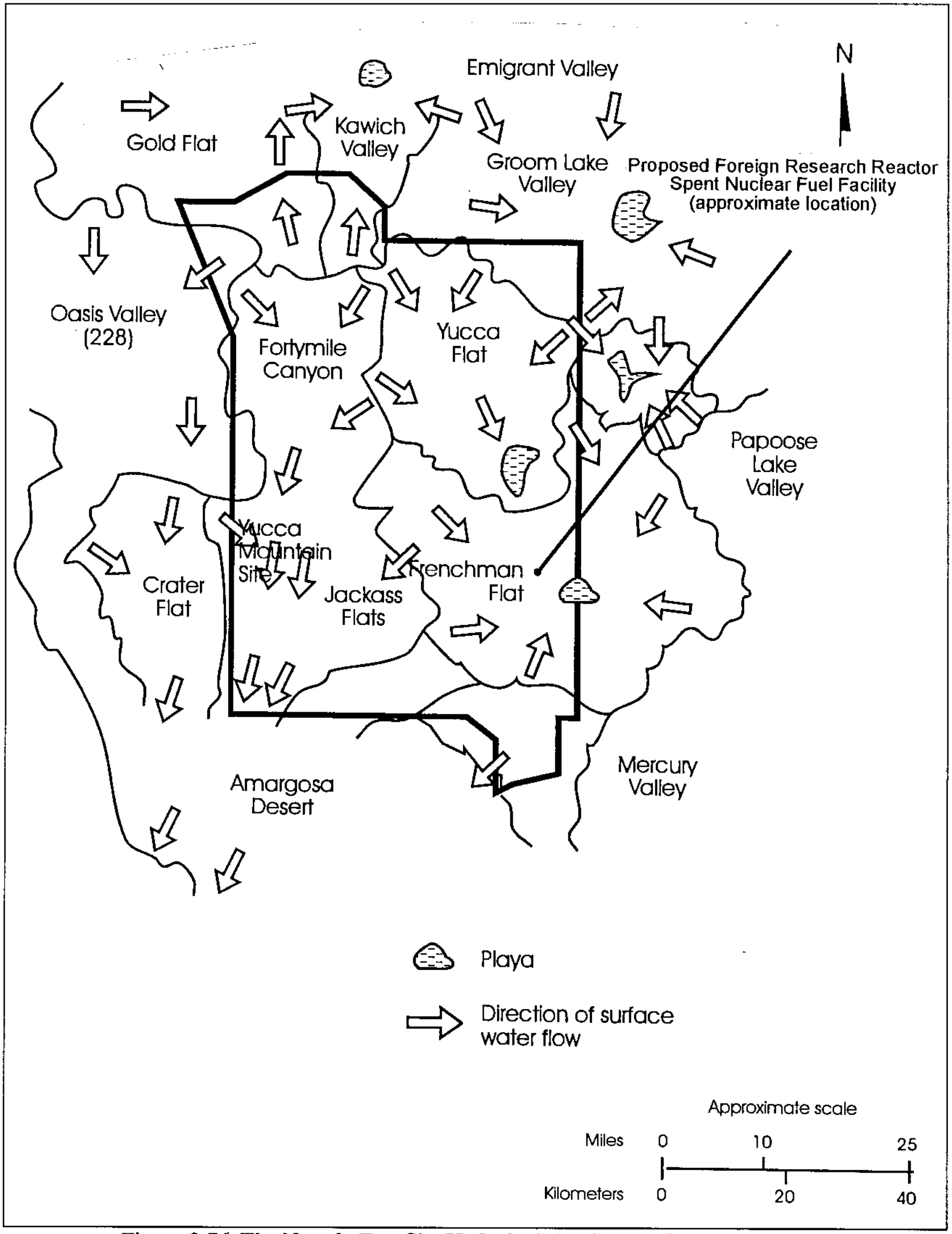


Figure 3-76 The Nevada Test Site Hydrologic Basins and Surface Drainage Direction

ephemeral, and therefore almost no streamflow data have been collected. The ephemeral character of streamflow has also limited the onsite monitoring of surface water quality. Perennial surface water originates from springs, and is restricted to source pools at some large springs. Because of the extreme aridity of this region, most of the spring discharge travels only a short distance before evaporating or infiltrating back into the ground (DOE, 1986b). The western half and southernmost part of the Nevada Test Site have channel systems which carry runoff beyond the Nevada Test Site boundaries during infrequent, very intense storms.

Two watersheds, Fortymile Canyon and Jackass Flats, have the potential for endangering offsite public health and safety due to flooding. Regional peak-flood flow equations for the southern Nevada area indicate that the 100-yr peak flow from the Fortymile Canyon drainage is approximately 370 m^3 per sec (97,744 gal per sec), and 230 m^3 per sec (60,760 gal per sec) from the Jackass Flats drainage (USAF et al., 1991). Underground nuclear testing has resulted in the release of radioactive materials at the land surface. There is the potential for 100-yr floods to transport these contaminants beyond the boundaries of the Nevada Test Site.

3.3.5.3.2 Groundwater

The hydrogeology at the Nevada Test Site is characterized by great depths to the groundwater table, and slow velocity of movement of water in the saturated and unsaturated zones (DOE, 1992c). Depth to groundwater varies from about 200 m (660 ft) beneath valleys in the southern part of the Nevada Test Site, to more than 500 m (1,650 ft) beneath Pahute Mesa. Locally, there are perched water tables at shallow depths (USAF et al., 1991). Perched aquifers have been reported at depths of 21 m (70 ft) in the southwestern part of Frenchman Flat. In the eastern portions of the Nevada Test Site, the water table occurs generally in the alluvium and volcanic rocks above the regional carbonate aquifer (DOE, 1993a).

Six major aquifers occur in the area. The hydrologic and geologic properties of these aquifers vary (DOE, 1988a). The lower carbonate and valley fill aquifers are the main sources of groundwater in the eastern part of the Nevada Test Site (DOE, 1986b). Four major aquitards (units tending to retard the flow of groundwater) have been reported in the area (DOE, 1986b).

Regional groundwater flow is from the north and northeast toward the regional discharge area near Ash Meadows in the Amargosa Desert (Figure 3-77). In the western portions of the area, the regional flow is from the northwest to the south and southwest. Groundwater recharge to the Ash Meadows Sub basin occurs primarily from precipitation over the mountainous areas in the northern, eastern, and southern portions of the basin (DOE, 1988a).

The hydrogeologic units that supply potable water to the Nevada Test Site have been classified as Class IIA (currently a source of drinking water) and IIB (potentially a source of drinking water), in accordance with the U.S. Environmental Protection Agency's guidelines for groundwater classification. No aquifers at the Nevada Test Site have been designated as sole-source aquifers.

Groundwater Quality: The quality of the Nevada Test Site groundwater is suitable for most purposes, and generally meets U.S. Environmental Protection Agency secondary standards for major cations and anions, and the primary standards for deleterious constituents. Contamination by radionuclides occurs below the water table as well as in the unsaturated zone above it. This contamination is a result of underground nuclear testing. The principal confirmed or suspected contaminants from these tests include various radionuclides (primarily tritium) and heavy metals.

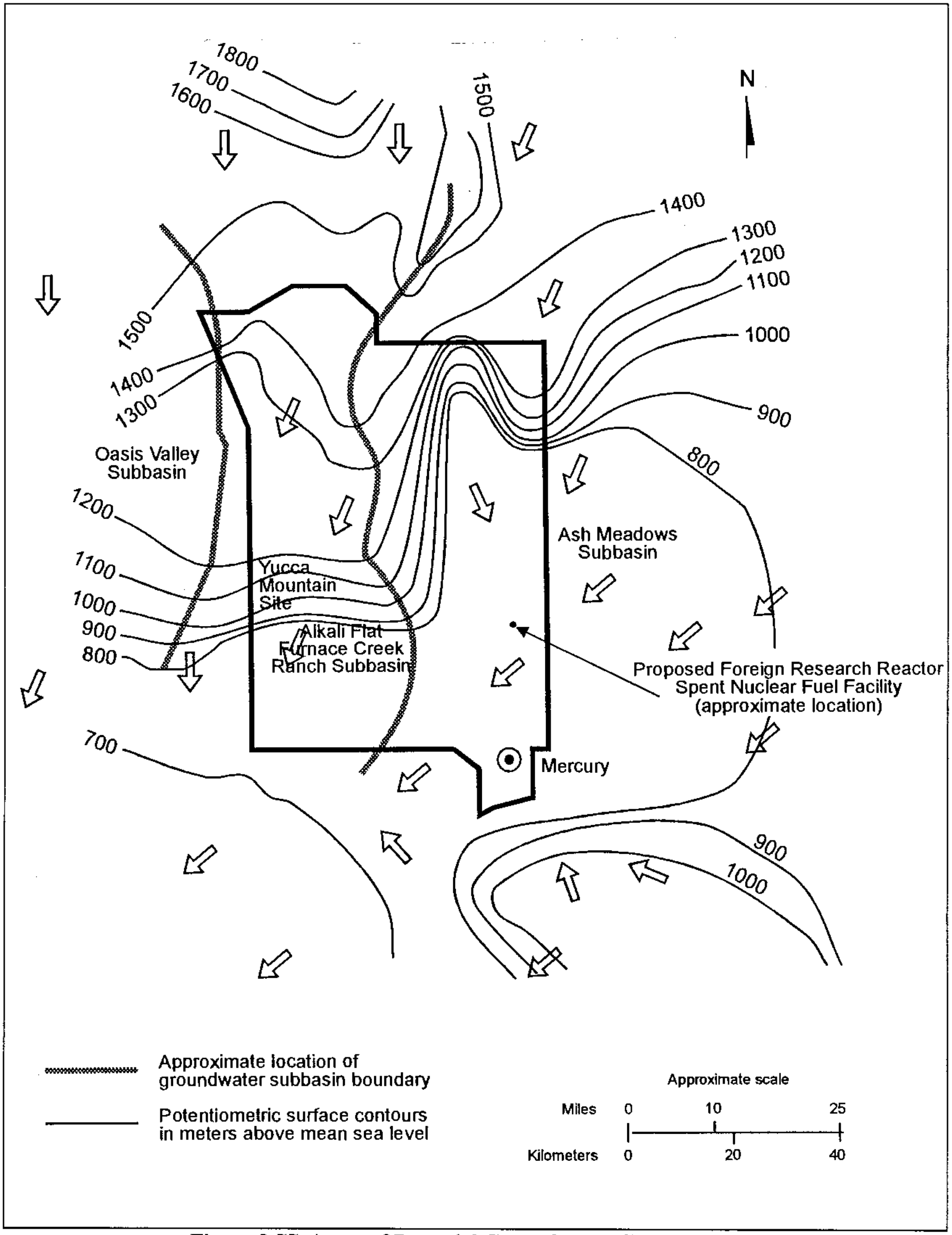


Figure 3-77 Areas of Potential Groundwater Contamination

Groundwater contamination could be transported toward the Nevada Test Site boundary by one of the regional groundwater flow systems. Groundwater flow velocities in these systems range between 1.8 and 183 m per year (6 and 600 ft per year). Due to sorption, most nuclides (other than tritium) would move at a much slower rate. The groundwater travel time from the Nevada Test Site to Ash Meadows Discharge Area is approximately 300 years. Radioactive decay, coupled with dilution and sorption, should reduce radioactivity to well below regulatory limits (USAF et al., 1991). Thus, there are no current effects on public health and safety, nor are any expected in the foreseeable future.

3.3.5.4 Meteorology

The climate at the Nevada Test Site and the surrounding region is characterized by high solar radiation, limited precipitation, low relative humidity, and large diurnal temperature ranges. The lower elevations have a climate typical of the Great Basin.

Wind: Low-level surface winds at the Nevada Test Site are influenced by the large-scale weather patterns interacting with the mountain ranges. Predominant winds are from the south during the summer, and the north during the winter. At Las Vegas, the peak wind gust on record is 145 km per hour (90 mph). Strong winds interacting with dry soil conditions are responsible for occasional duststorms or sandstorms.

Temperature and Humidity: At Area 6 (Figure 3-75) of the Nevada Test Site, the average daily maximum/minimum temperatures during the month of January are 10.6°C/-6.1°C (51.1°F/21.0°F). The average daily maximum/minimum temperatures are 35.6°C/13.9°C (96.1°F/57.0°F) in July. At Las Vegas, the coldest temperature on record is -13.3°C (8.1°F), and the warmest temperature on record is 46.7°C (116°F). The average relative humidity at 4 a.m. in Las Vegas is 40 percent. The average relative humidity at 4 p.m. is 20 percent (DOE, 1995c).

Precipitation: The average annual precipitation at Area 6 is 15 cm (5.9 in). Precipitation amounts for each month are generally less than 1.3 cm (.5 in). At Las Vegas, the greatest precipitation recorded in a 24-hr period is 6.6 cm (2.6 in). An average of 14 thunderstorm days occur each year, with maximum occurrence in July and August. Thunderstorms occasionally become severe. Tornadoes are extremely rare in Nevada (DOE, 1995c).

Atmospheric Dispersion: Data collected at Desert Rock for calendar year 1990 indicated that atmospheric conditions were unstable (Stability Classes A through C) approximately 25 percent of the time, neutral (Class D) approximately 37 percent of the time, and stable (Classes E through G) approximately 37 percent of the time for that year (DOE, 1995c).

Air Quality: In 1992, the majority of radioactive effluents at the Nevada Test Site originated from underground nuclear tests designed and conducted by two national laboratories and the Defense Nuclear Agency. Onsite monitoring of airborne particulates, noble gases, and tritiated water vapor indicated onsite concentrations that were generally not statistically different from background concentrations. Results of offsite environmental monitoring indicated none of the Nevada Test Site-related radioactivity was detected at any air sampling station, and there were no apparent net exposures detectable by the offsite dosimetry network (DOE, 1993a).

The nonradiological air emissions from the Nevada Test Site originate from concrete batch plants, aggregate crushing and processing, surface disturbance, fire training exercises, motor vehicle operations, boilers, and fuel storage. Based on the data collected by Engineering Science, Inc. at the ambient air monitoring stations, air quality at the Nevada Test Site is within applicable Federal and State standards (DOE, 1995c).

3.3.5.5 Ecology

The Nevada Test Site lies within the transition area of the Mojave desert and the Great Basin. The Nevada Test Site covers about 3,500 km^2 (1,350 mi^2), of which only 0.55 percent is developed (DOE, 1988b).

Plant communities on the Nevada Test Site have been classified according to the dominant shrub. Approximately 700 taxa have been identified on the Nevada Test Site (ERDA 1976; DOE, 1991b, DOE, 1993b). Figure 3-78 presents the general plant communities identified on the Nevada Test Site. The dominant plant communities in the Mojave desert are creosote bush. The dominant plant communities in the transition zone between the Mojave desert and the Great Basin are blackbrush, desert thorn, and hopsage. The dominant plant communities in the Great Basin are big sagebrush and black sagebrush, saltbush, and desert thorn.

There are more than 30 species of reptiles and amphibians, 190 species of birds, and 50 species of mammals on the Nevada Test Site (ERDA, 1976; RSN, 1993). Sewage ponds and man-made reservoirs have become an important resource for wildlife. Reptiles and amphibians on the Nevada Test Site include 1 species of desert tortoise, 14 species of lizards, and 17 species of snakes. Birds on the Nevada Test Site are often migratory and seasonal residents. The most-distributed species include the black-throated sparrow, house fin, red-tailed hawk, common raven, loggerhead shrike, mockingbird, ash-throated flycatcher, and mourning dove (Greger, n.d.a.; ERDA, 1976). The most abundant group of mammals on the Nevada Test Site are rodents.

There are several natural springs on the Nevada Test Site that feed flowing streams (Greger, n.d.a.). Vegetation along these channels consists of willow and tamarisk. National Wetlands Inventory maps are not available for the Nevada Test Site (DOE, 1995c).

Potential aquatic habitat on the Nevada Test Site includes surface drainage, playas, man-made reservoirs, and springs. Permanent surface water resources are limited to a few small springs. These surface drainage and playas are unable to support permanent fish populations (ERDA, 1976).

Threatened, Endangered, and Candidate Plant and Animal Species: Table 4.9-1 of Appendix F, Volume 1 of the SNF&INEL Final EIS presents a list of Federally and State-listed species that may be found in the vicinity of the Nevada Test Site (DOE, 1995c). There are no known plants that have been listed as threatened or endangered under the Endangered Species Act on the Nevada Test Site. However, the U.S. Fish and Wildlife Service has identified candidate species for listing, 11 of which may occur on or in the vicinity of the Nevada Test Site. Ten of these are Category 2 species (may be appropriate for listing as endangered or threatened but more information is needed).

Two listed reptile species on or in the vicinity of the Nevada Test Site are of concern. The chuckwalla is a Federal Candidate Category 2 species which may occur on the Nevada Test Site. The desert tortoise is the only Federally listed threatened species known to occur on the Nevada Test Site (DOE, 1995c).

Two bird species (American peregrine falcon and the bald eagle) which could occur on or within the vicinity of the Nevada Test Site are Federally listed endangered species. There are two (spotted bar and pygmy rabbit) Federal Candidate Category 2 mammal species identified as potentially occurring in the vicinity of the Nevada Test Site. There are no known fish species indigenous to the Nevada Test Site.

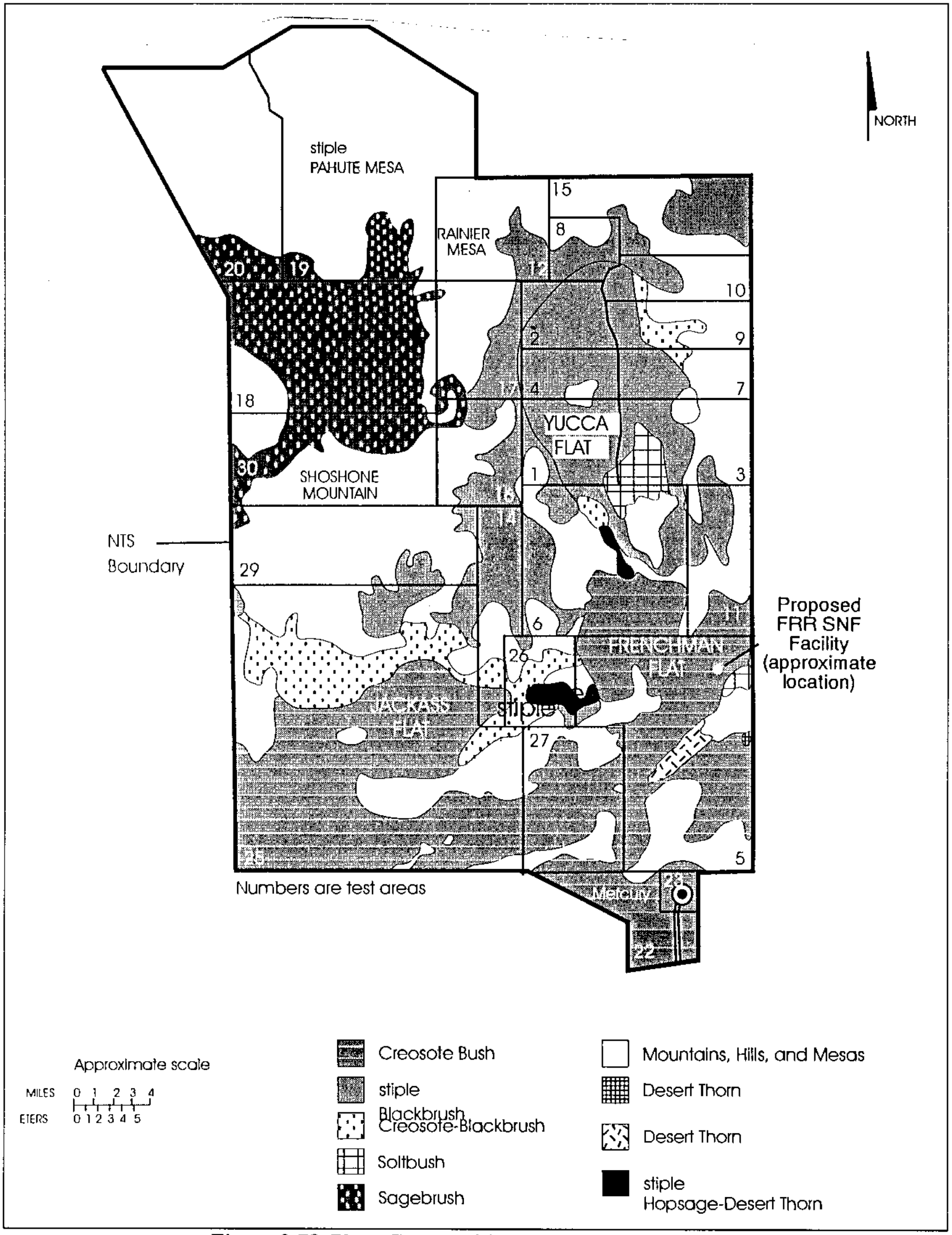


Figure 3-78 Plant Communities on the Nevada Test Site

3.3.5.6 Land Use

The Nevada Test Site occupies an area of approximately 3,500 km^2 (1,350 mi^2) in southern Nevada, in a sparsely populated area approximately 105 km (65 mi) northwest of Las Vegas. The Nevada Test Site is almost entirely surrounded by other Federally-owned lands that buffer it from lands open to the public. The Nevada Test Site is bordered by the Nellis Air Force Range on the north, east, and west, and by Bureau of Land Management lands on the south and southwest (DOE, 1993b).

Existing land use on the Nevada Test Site falls into four general categories: Testing Areas, Buffer/Reserved Areas, Industrial/Research Areas, and Waste Management Areas. According to the latest the Nevada Test Site land use map (Figure 3-79), approximately 50 percent of the land on the Nevada Test Site is buffer/reserved area for ongoing programs or projects (DOE, 1993a).

The Nevada Test Site is located in an area of sparsely vegetated desert. Principal uses in Nye County in the vicinity of the Nevada Test Site include mining, grazing, agriculture, and recreation (DOE, 1993a). Urban and residential land uses occur beyond the immediate vicinity of the Nevada Test Site. Clark County, to the southeast of the Nevada Test Site, consists of approximately 20,460 km^2 (7,900 mi^2), of which about 95 percent is owned by the Federal Government.

Numerous national, State, and local public recreation areas exist within the Nevada Test Site region. The Nevada Test Site is a controlled area, with public access limited to through traffic on U.S. Route 95, and on Lathrop Wells Road (DOE, 1993a). There are no onsite areas subject to Native American Treaty rights or that contain any prime or unique farmland (PIC, 1992).

3.3.5.7 Noise

The major noise sources at the Nevada Test Site occur primarily in developed operational areas, and include various facilities, equipment and machines, aircraft operations, and testing. No Nevada Test Site environmental noise survey data are available (DOE, 1995c). At the Nevada Test Site boundaries, noise from most sources is barely distinguishable from background noise levels. Transportation of people and materials to and from the Nevada Test Site is the noise source of importance to the public. During a normal work week about 3,300 employees travel to the Nevada Test Site each day (DOE, 1995c).

3.3.5.8 Transportation

Vehicular access to the Nevada Test Site is provided by U.S. Route 95 to the south, with off-road access to the northeast provided via Nevada State Route 375. Nevada State Route 375 and U.S. Route 95 are projected to remain at Level of Service A (free flow of traffic). The public transit serves the populated regions of Clark County. Contract buses run to the Nevada Test Site. There is no public transportation system serving the Nevada Test Site, but 70 buses a day transport employees to and from the site.

The nearest railroad is the Union Pacific, located approximately 80 km (50 mi) east of the Nevada Test Site near Las Vegas. No navigable waterways are capable of accommodating waterborne transportation of material shipments to the Nevada Test Site. McCarran International Airport in Las Vegas provides jet air passenger and cargo service from both national and local carriers.

3.3.5.9 Socioeconomics

A Nevada Test Site worker residential distribution survey from 1988 indicates that 88 percent lived in Clark County and 10 percent in Nye County (DOE, 1995c). In Clark County, most of the Nevada Test Site employees reside in the Las Vegas vicinity.

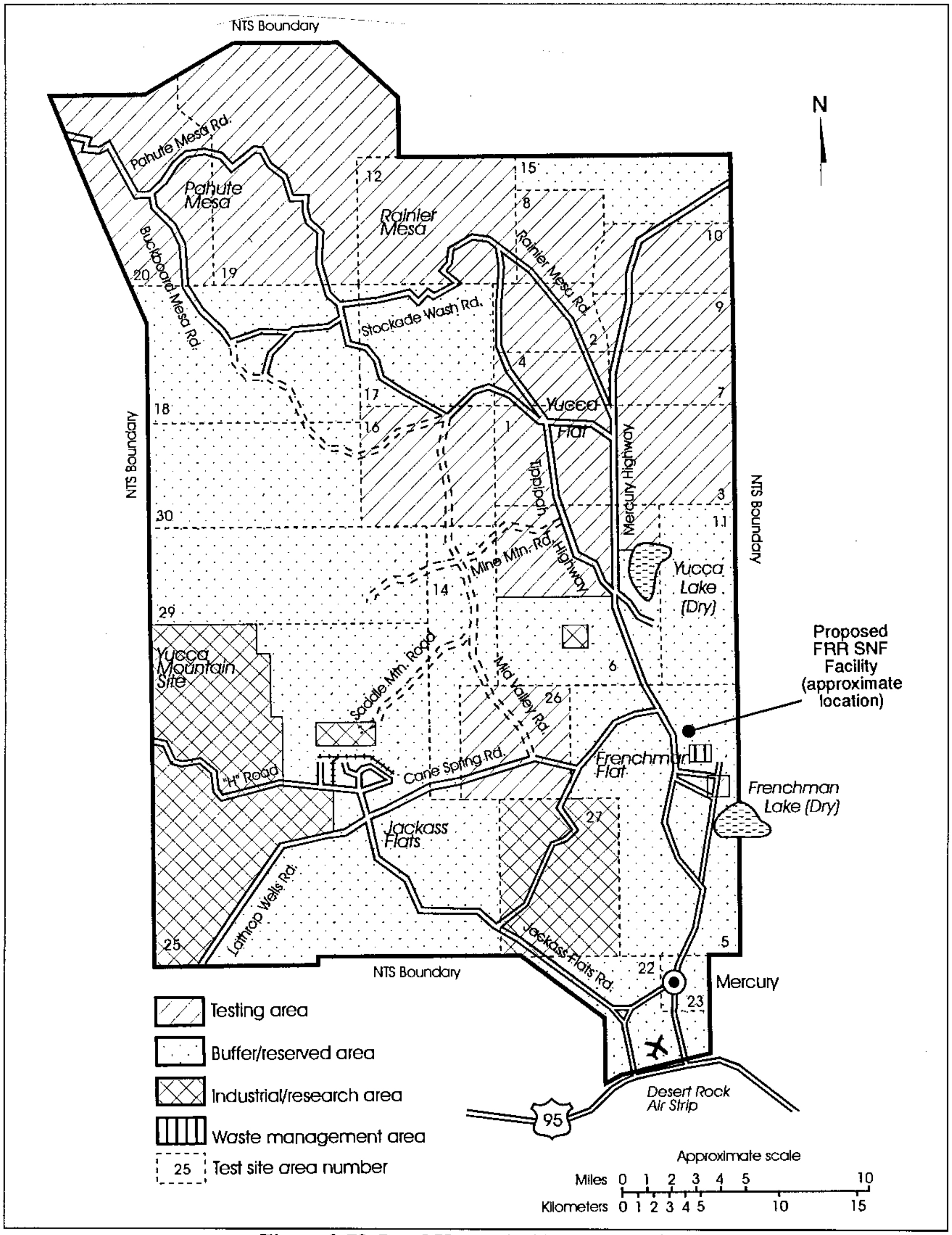


Figure 3-79 Land Use at the Nevada Test Site

Clark County: Clark County is composed of five incorporated cities (Las Vegas, Henderson, North Las Vegas, Boulder City, and Mesquite), and large expanses of unincorporated land. The area experiencing the majority of the county's development is the Las Vegas Valley (DOE, 1995c). Economic conditions in southern Nevada have improved continuously since the mid-1980's. The economy is driven by the growth in the hotel and gaming industry. Service employment accounts for nearly 45 percent of total employment, trade employment accounts for 21 percent, and Government and construction each account for an additional 10 percent (DOE, 1995c).

The unemployment rate reached a low of 4.9 percent in 1990, and increased to 7.5 percent as of June, 1993. However, the unemployment level is expected to decrease with new hotel, gaming, and amusement properties which opened at the end of 1993 (DOE, 1993a).

Nye County: The employment level in Nye county is low relative to Clark County, and includes opportunities in the services, mining, and Government sectors (DOE, 1993a). Nye County is sparsely populated, with the two largest population groupings in the communities of Pahrump and Tonopah. While tourist activity is an important part of the Nye County economy in communities along U.S. 95, mining is the major, even dominant, economic force.

The Nevada Test Site: The Nevada Test Site work force supports engineering design, construction, and operation of the site. As of January 1994, there were a total of 8,563 (3,286 on Nevada Test Site, 3,805 in Las Vegas, and 1,472 in the rest of Nevada or other areas) . The population within the 80 km (50 mi) radius of the Nevada Test Site is approximately 12,421. Minority population constitutes approximately 16 percent of the total. Figure 3-80 shows the racial and ethnic composition of the minority population within 80 km (50 mi) of the Nevada Test Site. Hispanics form more than 50 percent of the minority population.

The general characteristics of the low-income households residing within 80 km (50 mi) of the Nevada Test Site are presented in Figure 3-81. Low-income households are 48 percent of the total households.

The Nevada Test Site's fire protection capacity is structured to accommodate current mission requirements, with a self-contained firefighting department responsible for suppression and prevention. Other services include rescue, hazardous material response, training of fire personnel, fire prevention inspections, installation of all fire extinguishers at the Nevada Test Site, and fire prevention awareness programs. There is a mutual agreement between the Clark County Fire Department and all surrounding area departments to assist in any fire emergency when necessary (DOE, 1993a).

Health Care: The Nevada Test Site has a self-contained medical center that provides limited emergency treatment. Health care in the Las Vegas Area is provided through 13 full-service hospitals, with 3.44 hospital beds per 1,000 members of the population.

Education and Training: The Clark County school district provides education services for the employees who work at the Nevada Test Site. An average student/teacher ratio of 22:32 is reported for elementary school grades K-6 (DOE, 1993a). There are a number of vocational, training, and higher education institutions in the Las Vegas metropolitan area (DOE, 1993a).

Housing: Between 1980 and 1990, the number of housing units in Clark County increased by 84 percent. The increase in demand is attributable to the influx of retirees and other in-migrant population.

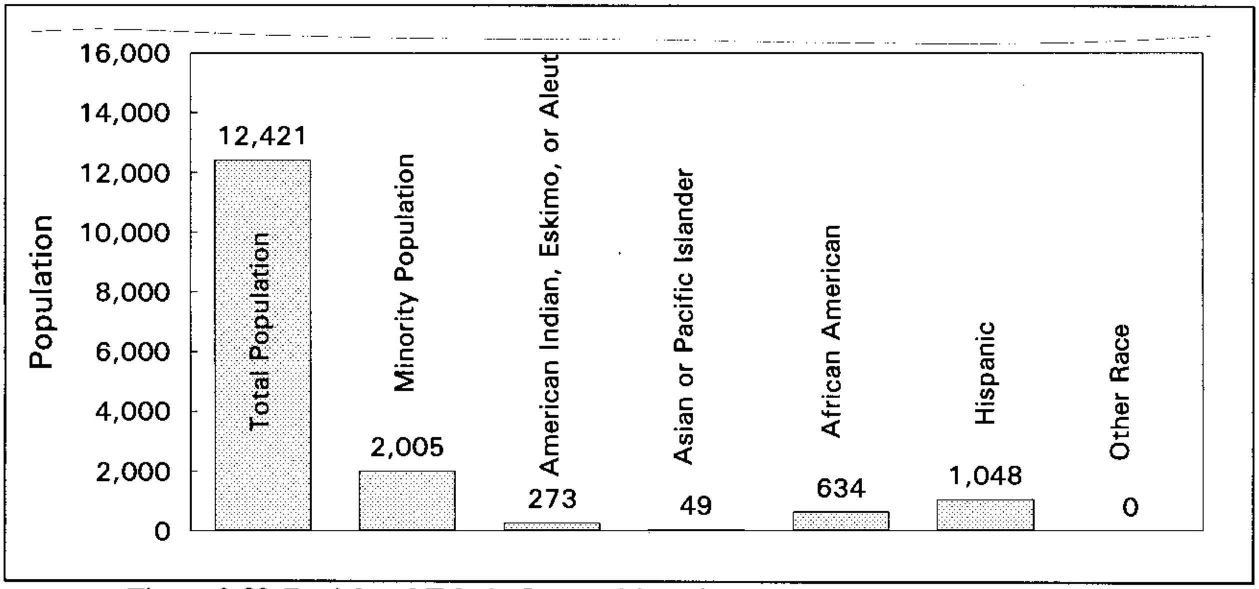


Figure 3-80 Racial and Ethnic Composition of the Minority Population Residing within 80 km (50 mi) of the Nevada Test Site

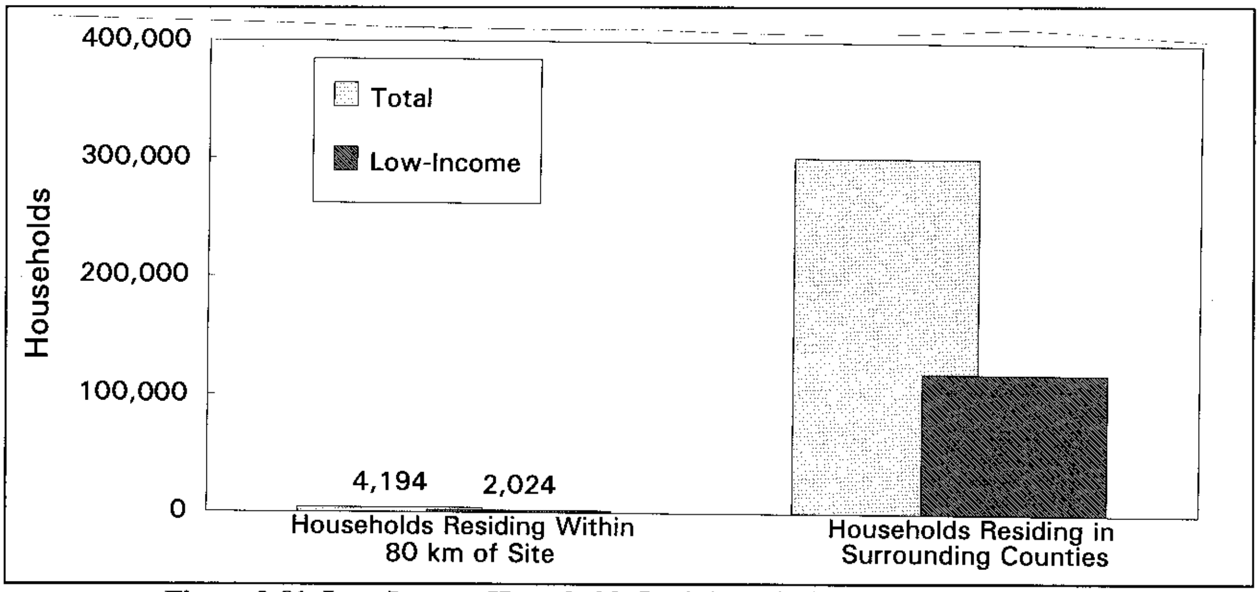


Figure 3-81 Low-Income Households Residing within 80 km (50 mi) of the Nevada Test Site

3.3.5.10 Historical, Archaeological, and Cultural Resources

People have inhabited the lands that comprise the Nevada Test Site for 12,000 years. The availability of the surface water was the primary determinant governing the location of past human occupation on these lands.

The Southern Paiute and Shoshone Native American tribes are known to have inhabited southern Nevada, including parts of what is now the Nevada Test Site. No known Native American resources are located on the Nevada Test Site (DOE, 1995c).

4. Policy Considerations and Environmental Impacts

This Chapter of the Environmental Impact Statement (EIS) describes the policy considerations and potential environmental impacts resulting from each of the management alternatives for implementation of the proposed action and the No Action Alternative. The environmental analysis addresses potential impacts of each alternative on workers, the public, and the environment. The general methodology used throughout this chapter is discussed in Section 4.1.

The policy considerations and environmental impacts of policy alternatives are described in this chapter. One policy alternative is the proposed action, which proposes the adoption of a policy whereby the United States would become involved in the management of the foreign research reactor spent nuclear fuel. The proposed action contains three separate management alternatives for adopting the policy. These management alternatives each contain different implementation alternatives related to that specific management alternative. The second policy alternative is the No Action Alternative which would involve no action by the United States in relation to the foreign research reactor spent nuclear fuel.

Each management alternative would result in very different policy considerations. Much of the foreign research reactor spent nuclear fuel analyzed in this EIS contains highly-enriched uranium (HEU), which can be used to make nuclear weapons. By adopting a policy to manage the foreign research reactor spent nuclear fuel, the proposed action would promote the U.S. goal of nuclear weapons nonproliferation by removing large amounts of HEU from civilian commerce. The No Action Alternative would be in direct conflict with the stated U.S. nuclear weapons nonproliferation goal and would seriously undermine credibility of the United States as a reliable partner in international nuclear weapons nonproliferation activities. Further, foreign research reactor operators may accuse the United States of failing to comply with its obligations under Article IV of the Non-Proliferation Treaty to share the benefits of peaceful nuclear cooperation with other countries.

Each management alternative would also result in very different environmental impacts in the United States which may vary according to the implementation alternatives of each management alternative. The No Action Alternative would have no direct environmental impacts in the United States.

Each of the three management alternatives under the proposed action is briefly summarized here. The three management alternatives were described in greater detail in Chapter 2, Sections 2.2 through 2.4. The policy considerations and environmental impacts of each alternative are described in detail in this chapter.

Management Alternative 1 — Manage Foreign Research Reactor Spent Nuclear Fuel in the United States

Management Alternative 1 of the proposed action entails acceptance and management of the foreign research reactor spent nuclear fuel in the United States. This management alternative would have direct environmental impacts in the United States.

Management Alternative 1 is composed of nine basic implementation components, as well as seven implementation alternatives that alter one of these basic components in some manner. The basic implementation of Management Alternative 1, as well as the seven implementation alternatives, are described in detail in Chapter 2, Section 2.2. The policy considerations and environmental impacts of the basic implementation of Management Alternative 1 are presented in Section 4.2. The policy considerations and environmental impacts of the seven implementation alternatives of Management Alternative 1 are presented in Section 4.3.

Management Alternative 2 — Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

Management Alternative 2 of the proposed action entails U.S. facilitation of overseas management of the foreign research reactor spent nuclear fuel at one or more foreign locations. No foreign research reactor spent nuclear fuel would be accepted into the United States. This would require advance negotiations and agreements with foreign reactor operators, officials in foreign governments, and reprocessing facilities. The outcome of these negotiations is uncertain. This management alternative would have no direct environmental impacts in the United States, unless the Department of Energy (DOE) decides to accept vitrified high-level waste from reprocessing facilities overseas in place of the foreign research reactor spent nuclear fuel. Very few countries have the capability to accept and store high-level wastes (GAO, 1994).

Management Alternative 2 is described in detail in Chapter 2, Section 2.3. Under this management alternative, the United States would negotiate some form of technical assistance and/or financial incentives in return for maintaining some measure of control over the foreign research reactor spent nuclear fuel containing U.S.-origin HEU. The policy considerations and environmental impacts of the two subalternatives of Management Alternative 2 are presented in Section 4.4.

Management Alternative 3 — Combination of Elements from Management Alternatives 1 and 2 (Hybrid Alternative)

Management Alternative 3 entails some combination of the elements from Management Alternatives 1 and 2, and is referred to as the Hybrid Alternative. Management Alternative 3 would likely have more direct environmental impacts in the United States than Management Alternative 2, but less than Management Alternative 1.

Management Alternative 3 is described in detail in Chapter 2, Section 2.4. For purposes of analysis, a sample Hybrid Alternative has been included to demonstrate one possible combination of elements within Management Alternatives 1 and 2, and to allow an analysis of its impacts. It is important to note that the Hybrid Alternative described is merely an example for analysis purposes, and is only one of numerous possible combinations of elements from Management Alternatives 1 and 2.

Under the Hybrid Alternative described, DOE and the Department of State would facilitate the reprocessing of the foreign research reactor spent nuclear fuel at western European reprocessing facilities (i.e., Dounreay, United Kingdom or Marcoule, France) for foreign research reactor operators in countries that can accept the reprocessing waste, as in Management Alternative 2. DOE would accept and manage the remaining foreign research reactor spent nuclear fuel in the United States, as in Management Alternative 1. The policy considerations and environmental impacts of the sample Hybrid Alternative (Management Alternative 3) are described in Section 4.5.

Other Alternatives and Comparisons

The No Action and Preferred Alternatives are discussed in Sections 4.6 and 4.7, respectively. Comparisons across all the alternatives of the potential impacts and costs are presented in Section 4.8 and 4.9, respectively. Finally, this chapter concludes by comparing the risks due to the alternatives to the risks due to other common activities in Section 4.10.

4.1 Overview of Environmental Impacts

4.1.1 Presentation of the Environmental Impacts

Potential environmental impacts associated with each segment of the affected environment of the proposed action are addressed in this chapter. These segments are presented in this section in the following order:

- Marine transport impacts,
- Port of entry impacts,
- Ground transport impacts, and
- Management Site impacts.

The impact analyses of these four segments are described in more detail in Appendices C, D, E, and F, respectively. Effects of each implementation alternative of Management Alternative 1 of the proposed action on U.S. nuclear weapons nonproliferation goals and objectives are also discussed. In addition, this chapter summarizes the potential costs associated with the alternatives. Details on costs are presented in Appendix F.

Spent nuclear fuel is transported in strong, heavy casks (NRC, 1993). After the spent nuclear fuel is delivered, the empty casks must be transported back on a return trip. Under most of the alternatives, empty casks would be transported overland, through U.S. ports, and on ships. There would be minor nonradiological impacts (vehicle emissions and potential traffic accidents) during ground transport of empty casks. These nonradiological ground transport impacts are included as part of the assessment in this EIS.

4.1.2 Key Assessment Factors

A key assessment factor is one that may differentiate among alternatives, has a measurable impact, or be of public interest. The detailed analysis of potential environmental impacts presented in the appendices of this EIS did not reveal any factor likely to cause a large impact. Because radiation exposure and its consequences is a topic of great public interest, emphasis is placed upon exposure to radiation, although DOE considers the evaluated effects of radiation to be small.

During handling operations, the principal hazard would come from radiation being emitted by the foreign research reactor spent nuclear fuel. Without adequate shielding, the radiation levels at the surface of some of the spent nuclear fuel itself would often be high enough to induce a prompt fatality. This radiation can and would be attenuated (i.e., reduced) by the shielding materials of the transportation cask, such as lead, steel, and polyethylene. Further, since radiation intensity decreases with distance, maintaining a distance from the cask would also provide radiation protection. At 100 m (330 ft) from the cask, the radiation levels would not be detectable above background radiation. All foreign research reactor spent nuclear fuel handling at the proposed foreign research reactor spent nuclear fuel management sites would take place at considerable distances from the public (greater than 100 m or 330 ft). Recently, actual radiation measurements were taken by the State of North Carolina, Department of Environment, Health, and Natural Resources, of the casks used in the first shipment of the 153 spent fuel elements covered by the Urgent Relief Environmental Assessment (DOE, 1994m). In every case, the State of North Carolina reported detecting no radiation above background levels (radiation exposure from natural sources) at a distance of 1 meter (3.3 ft) from the package surface (State of North Carolina, 1994).

Accidents involving foreign research reactor spent nuclear fuel could potentially also result in releases of radioactive material which could cause radiation exposures. For most accidents, essentially none of the radioactive material would be released because it is an integral part of the solid fuel. Larger quantities of radioactive elements could be released only when the accident generates enough energy to release particles of foreign research reactor spent nuclear fuel to the atmosphere, such as with a fire. However, the probability of such accidents is very small. For most accidents, the energy would not be high enough to damage the foreign research reactor spent nuclear fuel, so that none of the radioactive material would be released.

4.1.3 General Radiological Health Effects

The effect of radiation on people depends upon the kind of radiation exposure (alpha and beta particles, and gamma and x-rays) and the total amount of tissue exposed to radiation. The amount of radiant energy imparted to tissue from exposure to ionizing radiation is referred to as absorbed dose. The sum of the absorbed dose to each tissue, when multiplied by certain quality and weighting factors that take into account radiation quality and different sensitivities of these various tissues, is referred to as effective dose equivalent (EDE).

An individual may be exposed to radiation from outside the body, or from inside the body because radioactive materials may enter the body by ingestion or inhalation. External dose is different from internal dose in that it is delivered only during the actual time of exposure. An internal dose, however, continues to be delivered as long as the radioactive source is in the body (although both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time). The dose from internal exposure is calculated over 50 years following the initial exposure.

The annual radiation dose limit to the public from nuclear facilities operated by DOE is 100 mrem per year (NRC, 1991). The potential foreign research reactor spent nuclear fuel management sites covered by DOE operations normally operate such that the public's dose is undetectable. For comparison, it is estimated that the average individual in the United States receives a dose of about 350 mrem per year from all sources combined, including natural and medical sources of radiation and radon. A modern chest x-ray, for example, results in an approximate dose of 8 mrem, while a diagnostic hip x-ray results in an approximate dose of 83 mrem (DOE, 1995c).

Radiation can also cause a variety of adverse health effects in people. A large dose of radiation can cause prompt death. At low doses of radiation, the most important adverse health effect for depicting the consequences of environmental and occupational radiation exposures (which are typically low doses) is the potential inducement of cancers that may lead to death in later years. This effect is referred to as latent cancer fatalities (LCF) because the cancer may take years to develop and for death to occur, and may never actually be the cause of death.

In addition to LCF, other health effects could result from environmental and occupational exposures to radiation. These effects include nonfatal cancers among the exposed population and genetic effects in subsequent generations. Table 4-1 shows the dose-to-effect factors for these potential effects as well as for LCF. For simplicity, this EIS presents estimated effects of radiation only in terms of LCF. The nonfatal cancers and genetic effects are less probable consequences of radiation exposure, and are less serious.

Table 4-1 Risk of LCF and Other Health Effects from Exposure to Radiation

Population[a]	*LCF*[b]	*Nonfatal Cancers*	*Genetic Effects*	*Total Detriment*
Workers	0.0004	0.00008	0.00008	0.00056
Public	0.0005	0.0001	0.00013	0.00073

[a] *The difference between the worker risk and the general public risk is attributable to the fact that the general population includes more individuals in sensitive age groups (that is, less than 18 years of age and more than 65 years of age).*

[b] *When applied to an individual, units are lifetime probability of LCF per rem of radiation dose. When applied to a population of individuals, units are excess number of cancers per person-rem of radiation dose. Genetic effects as used here apply to populations, not individuals.*

The collective or "population" dose to an exposed population is calculated by summing the estimated doses received by each member of the exposed population. This is referred to as a "population dose." The total population dose received by the exposed population is measured in person-rem. For example, if 1,000 people each received a dose of 0.001 rem, the population dose would be 1.0 person-rem (1,000 persons x 0.001 rem = 1.0 person-rem). The same population dose (1.0 person-rem) would result if 500 people each received a dose of 0.002 rem (500 persons x 0.002 rem = 1 person-rem).

The factor used in this EIS to relate a dose to its effect is 0.0004 LCF per person-rem for workers and 0.0005 LCF per person-rem for individuals among the general population (DOE, 1995c). The latter factor is slightly higher because of some individuals in the public, such as infants, who may be more sensitive to radiation than workers. These factors are based on the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP, 1991), and are consistent with those used by the U.S. Nuclear Regulatory Commission (NRC) in its rulemaking *Standards for Protection Against Radiation* (NRC, 1991). The factors apply where the dose to an individual is less than 20 rem and the dose rate is less than 10 rem per hour. At doses greater than 20 rem, the factors used to relate radiation doses to LCF are doubled. At much higher doses, prompt effects, rather than LCF, may be the primary concern. Unusual accident situations that may result in high radiation doses to individuals are considered special cases. No such cases are expected with either incident-free handling or accidents with foreign research reactor spent nuclear fuel.

These concepts may be applied to estimate the effects of exposing a population to radiation. For example, if 100,000 people were each exposed only to background radiation (0.3 rem per year), 15 LCF per year would be expected (100,000 persons x 0.3 rem per year x 0.0005 LCF per person-rem = 15 LCF per year).

Sometimes, calculations of the number of LCF associated with radiation exposure do not yield whole numbers and, especially in environmental applications, may yield numbers less than 1.0. For example, if 100,000 people were each exposed to a total dose of only 1 mrem (0.001 rem), the population dose would be 100 person-rem, and the corresponding estimated number of LCF would be 0.05 (100,000 persons x 0.001 rem x 0.0005 LCF per person-rem = 0.05 LCF).

The *average* number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people is 0.05. In most groups, nobody (zero people) would incur an LCF from the one mrem dose each member would have received. In a small fraction of the groups, one latent fatal cancer would result; in exceptionally few groups, two or more latent fatal cancers would occur. The average number of deaths over all the groups would be 0.05 latent fatal cancers (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome is zero LCF.

These same concepts apply to estimating the effects of radiation exposure on a single individual. Consider the effects, for example, of exposure to background radiation over a lifetime. The "number of LCF" corresponding to a single individual's exposure to 0.3 rem per year over a (presumed) 72-year lifetime is:

> 1 person x 0.3 rem per year x 72 years x 0.0005 LCF per person-rem = 0.011 LCF or one chance in 91 of an LCF.

Again, this should be interpreted in a statistical sense; that is, the estimated effect of background radiation exposure on the exposed individual would produce a 1.1 percent chance that the individual would incur a latent fatal cancer. Alternatively, this method estimates that about 1 person in 91 would die of cancers induced by background radiation.

4.1.4 Risks

Another concept important to the presentation of results in this EIS is the concept of risk. Risks are most important when presenting accident analysis results. The chance that an accident might occur during the conduct of an operation is called the probability of occurrence. An event that is certain to occur has a probability of 1.0 (as in 100 percent certainty). If an accident is expected to happen once every 50 years, the frequency of occurrence is 0.02 per year (1 occurrence every 50 years = 0.02 occurrences per year). A frequency estimate can be converted to a probability statement. If the frequency of an accident is 0.02 per year, the probability of the accident occurring in a 10-year program is 0.2 (10 years x 0.02 occurrences per year).

Once the frequency (occurrences per year) and the consequences (for radiation effects, measured in terms of the number of LCF caused by the radiation exposure) of an accident are known, the risk can be determined. The risk per year is the product of the annual frequency of occurrence times the number of LCF. This annual risk expresses the expected number of LCF per year, taking account of both the annual chance that an accident might occur and the estimated consequences if it does occur.

For example, if the frequency of an accident were 0.2 occurrences per year and the number of LCF resulting from the accident were 0.05, the risk would be 0.01 LCF per year (0.2 occurrences per year x 0.05 LCF per occurrence = 0.01 LCF per year). Another way to express this risk (0.01 LCF per year) is to note that if the operation subject to the accident continued for 100 years, one LCF would be likely to occur because of accidents during that period. This is equivalent to 1 chance in 100 that a single LCF would be caused by the accident source for each year of operation. This risk can be related to the risk of death from other accidental causes for comparison. As an example, the risk of dying from a motor vehicle accident is about 1 chance in 80. Similarly, the risk of death for the average American from fire is approximately 1 chance in 500, and for death from accidental poisoning, the risk is about 1 chance in 1,000 (NNPP, 1993). Section 4.10 compares the risks calculated in this EIS to those of common activities.

4.1.5 Estimated Radiation Dose Rate Near the Foreign Research Reactor Spent Nuclear Fuel Transportation Casks

The regulatory external radiation dose rate limit for foreign research reactor spent nuclear fuel transportation casks selected for use in the marine and ground transport analysis is 10 mrem per hour at 2 m (6.6 ft) from the "exclusive use" vehicle (no other cargo) [49 Code of Federal Regulations (CFR) 173.441]. This is equivalent to approximately 23 mrem per hour at 1 m (3.3 ft). Historical data from actual cask shipments of research reactor spent nuclear fuel have shown dose rates considerably below this regulatory limit. Dose measurements of casks containing research reactor spent nuclear fuel, including the foreign research reactor spent nuclear fuel recently received under the Urgent Relief Environmental

Assessment (DOE, 1994m), are presented in Appendix F, Section F.5. The average of these measurements is 2.3 mrem per hour at 1 m (3.3 ft) from the surface of the cask. Recent measurements taken by the State of North Carolina on foreign research reactor spent nuclear fuel shipment packages, covered by the Urgent Relief Environmental Assessment, showed that the external dose rate at 1 m (3.3 ft) was undetectable above background radiation levels (State of North Carolina, 1994).

To be conservative, the analyses in this chapter use the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the side of the transport vehicle for the radiation dose rate near the foreign research reactor spent nuclear fuel casks. This conservative value was used in the calculations of incident-free doses to members of the public, marine transport workers, port workers, and ground transport workers. For radiation workers at the spent nuclear fuel management sites, the dose rate in the vicinity of the casks was estimated by the conservative methodology presented in Appendix F, Section F.5.

4.1.6 The Effects of Radiation on Plants and Animals

There is no convincing evidence from the scientific literature that chronic radiation doses below 1 rad per day will harm animal or plant populations. It is highly probable that limitation of the exposure of the most exposed humans (the critical human group, living on and receiving full sustenance from the local area) to 100 mrem per year will lead to dose rates to plants and animals in the same area of less than 1 rad per day. DOE and NRC regulations limit annual human exposures to values far lower than those that have caused observable damage in plant and animal populations. Therefore, specific radiation protection standards for nonhuman biota are not needed (IAEA, 1992).

4.2 Management Alternative 1 – Manage Foreign Research Reactor Spent Nuclear Fuel in the United States – Basic Implementation

This section presents the policy considerations and potential environmental impacts of the basic implementation of Management Alternative 1. Under the basic implementation of Management Alternative 1, all the foreign research reactor spent nuclear fuel could be accepted into the United States. DOE and the Department of State believe this would promote the nuclear weapons nonproliferation objective of reducing, and ultimately eliminating, civil commerce in HEU. The spent nuclear fuel could be managed safely and securely at any of five DOE sites.

Policy Considerations

A critical result of this basic implementation of Management Alternative 1 would be the continued viability and vitality of the Reduced Enrichment for Research and Test Reactors (RERTR) Program, which has the goal of minimizing and eventually eliminating the use of HEU in civil nuclear programs. The successful development of alternative fuels for research reactors and the expansion of the program to Russia, the other Newly Independent States, China, South Africa, and other countries, and the establishment of a world-wide norm discouraging the use of HEU, is dependent on a United States commitment to action. Finally, this basic implementation of Management Alternative 1 would support the Administration's nuclear weapons nonproliferation objective of not encouraging reprocessing for either nuclear power or nuclear explosive purposes.

Another crucial consideration associated with Management Alternative 1 is the *Treaty on the Non-Proliferation of Nuclear Weapons*. The parties to the Non-Proliferation Treaty met in May of 1995 and agreed to extend the treaty indefinitely and without conditions. One key to the success of the 1995 Non-Proliferation Treaty Conference was the ability of the United States to convince other

Non-Proliferation Treaty parties that the nuclear weapons states had complied with their obligations under Article IV of the Non-Proliferation Treaty to assist the non-nuclear weapons states with peaceful applications of nuclear energy.

Although the Non-Proliferation Treaty was extended indefinitely, the parties also agreed to review the treaty every five years to ensure that all parties are in compliance. Any country which has been compelled to shut down its research reactors could accuse the United States of not having complied with its treaty obligations. This accusation, however ill-founded, could be made not only by the affected countries, but by any country opposed to the interests of the United States.

The amount of foreign research reactor spent nuclear fuel that would be accepted under the basic implementation of Management Alternative 1 is up to approximately 19.2 metric tons of heavy metal (MTHM) representing approximately 22,700 elements. This amount is an upper limit because if some nations were to reprocess their research reactor spent nuclear fuel, for example, the amount of foreign research reactor spent nuclear fuel accepted into the United States would be reduced. Under the basic implementation of Management Alternative 1, approximately 4.6 metric tons (5.1 tons) of HEU would be removed from international commerce.

4.2.1 Marine Transport Impacts

Because the basic implementation of Management Alternative 1 involves ocean transport, DOE and the Department of State considered the environmental impacts on the global commons (i.e., portions of the ocean not within the territorial boundary of any nation) in accordance with Executive Order 12114 (U.S. Federal Register, 1979).

4.2.1.1 General Assumptions and Analytic Approach

The basic implementation of Management Alternative 1 includes the shipment of approximately 837 transportation casks containing foreign research reactor spent nuclear fuel over a 13-year period. Of these, approximately 721 transportation casks would be transported by sea to the United States, with the remainder (116) coming overland from Canada. DOE would prefer to consolidate the approximately 721 casks on board ships to minimize the number of voyages, but it is also possible that approximately 721 voyages could be required. This section evaluates the impacts of the marine transportation, including shipment in international waters from the port of origin to the United States and coastal shipping in United States territorial waters.

Four types of commercial cargo ships are considered to be candidates to carry foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1: containerized, breakbulk (general cargo), roll-on/roll-off, and purpose-built vessels (see Appendix C for a more complete description of these vessels). DOE and the Department of State assumed that all casks would be transported in standard International Standards Organization 20-ft shipping containers, because this is current shipping practice.

Nonradiological impacts associated with the marine shipment of 721 containerized transportation casks would be minimal. The United States receives more than 56,000 ships engaged in foreign trade at its ports each year (DOC, 1994). Shipping an additional 56 containers per year on average over the 13-year receipt period is not likely to cause any additional ships to sail beyond the number already scheduled. In the event that chartered vessels are used for this program, up to 10 voyages per year could be required, which is only 0.02 percent of the number engaged in regular commerce. Additional nonradiological impacts would be very small whether chartered or regularly scheduled commercial vessels are used. The number of containers handled on a regular basis is so large that the addition of the foreign research reactor spent nuclear fuel containers would add essentially no impacts (cargo vessels typically carry 800 to 1,000 containers per voyage). While nonradiological marine events such as unloading or cargo shifting accidents would be possible, the nonradiological impacts would be miniscule.

The radiological impacts of transporting the foreign research reactor spent nuclear fuel by sea were considered in two ways, incident-free impacts and accident impacts. The incident-free impacts would be those that occur simply due to the marine shipping of foreign research reactor spent nuclear fuel, assuming there are no accidents. The ship's crew would be the affected individuals in this case. The accident impacts would be the consequences of reasonably foreseeable accidents that might occur. These two evaluations are discussed in the following two sections, with additional details in Appendix C.

4.2.1.2 Conservative Assumptions and Maximum Estimated Impacts of Incident-Free Marine Transport

The primary impact of incident-free marine shipping of foreign research reactor spent nuclear fuel would be upon the crews of the ships used to carry the spent nuclear fuel casks. Since the crew of a ship is normally separated from the cargo and shielded by both the cargo and the ship's structure, the risk to the crew from spent nuclear fuel transport during most crew activities would be extremely low (DOE, 1994m). The exceptions would include the exposure to the crew during loading and off-loading of the spent nuclear fuel ISO containers and during daily inspection of the ship's cargo, including the containers housing the spent nuclear fuel transportation casks. Therefore, the crew exposure during loading, daily inspection, and unloading of the transportation casks has been incorporated into the incident-free marine transport analysis. The exposure to dock workers at the foreign research reactor spent nuclear fuel port of entry is assessed in Section 4.2.2.

Daily inspections of the casks is the activity that would result in the largest doses to the ship's crew, with the inspectors considered the maximally exposed workers during incident-free marine transport. For any given voyage, DOE and the Department of State conservatively assumed that the same three inspectors would conduct all of the inspections. The impact on the inspectors would be a function of the number of inspections performed, which would depend upon the amount of time the cask is onboard. Therefore, the incident-free radiological impact on the inspectors would depend upon the total duration of the voyage, including days at sea, in intermediate ports, and days in coastal sailing between intermediate ports. The duration of the voyage was selected as the weighted average of the duration of all the shipments necessary for 721 transportation casks. (See Appendix C for further details regarding this assumption.)

To maximize the estimated impact from incident-free transport, DOE and the Department of State made conservative assumptions regarding crew exposure. Specifically, DOE and the Department of State conservatively assumed that eight and two casks (loaded two casks per hold) would be shipped per voyage of chartered and regularly scheduled commercial ships, respectively. This assumption would result in additional exposure of the ship's crew due to the effect of loading casks into holds where a loaded cask would have already been stowed, and would also increase the exposure to the crew members performing daily inspections. The additional exposure would be a result of the combination of the radiation fields surrounding each of the transportation casks.

Assuming 56 casks per year, the number of annual voyages required would range from 7 to 28, depending upon the number of casks per ship. Although the foreign research reactor spent nuclear fuel would be shipped from 40 countries worldwide and to both U.S. coasts over a 13-year receipt period, DOE and the Department of State conservatively assumed that a single crew could be involved in up to 9 voyages per year. As a practical matter, this overstates the rate at which a crew would sail from Europe or Asia and back. Additionally, to determine the dose to the maximally exposed worker in the ship's crew, DOE and the Department of State conservatively assumed that the same individuals would conduct all the daily onboard inspections.

The dose received during daily cargo inspection would be a major contributor to the crew dose, so the duration of the voyage is an important consideration. Chartered vessels would sail directly to the port(s) of entry, yielding an average voyage duration of 18 days. DOE and the Department of State conservatively assumed that all shipments aboard regularly scheduled commercial breakbulk vessels would include two intermediate port stops in the United States, which would add 3 days to the voyage.

Table 4-2 presents the maximum estimated incident-free marine transport doses and risks. Values are provided for a chartered ship (which would not make intermediate port calls) and for a regularly scheduled commercial vessel. The values are based on the estimated time the cask would be onboard multiplied by the dose per day received as a result of inspections, plus the crew dose due to the foreign research reactor spent nuclear fuel container loading and off-loading activities. While the use of a chartered ship would result in higher per-shipment impacts (eight casks per shipment versus two for regularly scheduled commercial ships), the reduced number of voyages would offset this increase in per-shipment impacts. Therefore, the use of chartered ships instead of regularly scheduled commercial ships would result in slightly lower total crew exposures in the basic implementation of Management Alternative 1. The selection of the shipping mode, however, would not be based on crew exposures alone. Other factors, such as cost, would also be important in the choice of chartered or regularly scheduled ships. The results in Table 4-2, therefore, provide an estimate of the range of maximum worker exposures due to the shipment of the foreign research reactor spent nuclear fuel.

Table 4-2 Incident-Free Marine Transport Impacts[a]

	Regularly Scheduled Commercial Ship				*Chartered Ship*			
	Maximally Exposed Worker Dose (mrem)	***Maximally Exposed Worker Risk (LCF)***	***Population Dose to Crew (person-rem)***	***Population Risk (LCF)***	***Maximally Exposed Worker Dose (mrem)***	***Maximally Exposed Worker Risk (LCF)***	***Population Dose to Crew (person-rem)***	***Population Risk (LCF)***
Impacts Per Shipment	66[b]	0.000027	0.23	0.000091	100[b]	0.00004	0.83	0.00033
Impacts for the Basic Implementation	1,300[b,c]	0.00052[c]	85	0.034	1,300[b,c]	0.00052[c]	75	0.030

[a] *These results are based on the assumption that the dose rates associated with the casks are all derived from the exclusive-use regulatory limit. Historically, the average of these dose rates has been equal to about one-tenth of this regulatory limit, so this assumption is conservative.*

[b] *If an individual works on repeated shipments, this maximally exposed worker dose could exceed the annual regulatory limit. Therefore, DOE would require that mitigation measures be implemented to keep the maximally exposed worker dose down to 100 mrem per year or lower. See Appendix C for estimates of the total exposure to the ships' crews without mitigation measures.*

[c] *These results are based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years.*

Marine transport workers are not trained to be radiation workers, so they would not be subject to the radiation worker limit of 5,000 mrem/yr. The applicable regulatory limit for these workers would be the same as for the general public: 100 mrem/yr. As the table shows, the highest estimated maximally exposed worker risk is 0.00052 LCF, which is based on the annual regulatory limit every year for 13 years. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

The highest estimated population risk is about 0.034 LCF, which is much less than 1 LCF.

4.2.1.3 Conservative Assumptions and Maximum Estimated Impacts of Accidents During Marine Transport

The basic implementation of Management Alternative 1 could potentially impact the marine environment in the event of an accident involving the release of radioactive material from the spent nuclear fuel. This section discusses possible accidents and their consequences.

The range of accidents that could occur during marine transport is quite broad. The ship could collide with another vessel or an object such as a shoal, rock, or wreck. Foul weather could damage or sink the ship, or the ship could experience a fire, explosion, or other problem. To reduce the risk due to potential accidents, the casks that would carry the foreign research reactor spent nuclear fuel have been designed to prevent damage to the cask contents in all but the most severe, and least likely, cases. See Appendix B of this EIS for a description of the foreign research reactor spent nuclear fuel transportation casks.

Two scenarios emerge that could potentially threaten the marine environment and possibly humans: the cask could be damaged and then involved in a fire, or the cask could sink. These cases are discussed in more detail below.

Cask Damaged Followed by a Fire

A ship carrying foreign research reactor spent nuclear fuel could be involved in a severe collision with another ship. It is possible that a transportation cask, carried on a ship involved in such a collision, could be exposed to impact forces resulting from the collision. In that event, the cask could be damaged. However, only a small fraction, at most, of the force generated in a collision of one ship with another would be brought to bear on a transportation cask for two reasons. First, the force of a ship-to-ship collision would be distributed over the entire area of contact between the two ships, which means that the force density (force per unit of area) that would result from a collision must be considered. The maximum cross sectional area presented by a transportation cask would be small in comparison to the typical impacted area, so that even if a cask were located directly in the path of the collision and unprotected by intervening hulls, bulkheads, etc., the force that might be exerted on such a cask would be limited by the force density.

Secondly, ships floating on water are yielding objects, so that some portion of the energy of impact would be transmitted to the water. Even severe collisions with large impact forces, by themselves, would not necessarily result in catastrophic failure of a transportation cask. Thus, it would be even more unlikely for a less severe collision to result in the breach of a cask and, thereafter, a release of any of its contents. Attachment 4 to Appendix D discusses in detail the forces involved in ship collisions.

If it is assumed, however, that a ship collision breaches the cask, a release of radioactive material would be possible. In such a circumstance, the release would be small because the spent fuel is metallic and thus would release very limited quantities of radioactive material, even if mechanically damaged. However, due to the severity of the collision required to breach the cask, the ship carrying the foreign research reactor spent nuclear fuel cask would be severely damaged and probably would sink. Whether the ship would sink or not, the only humans that could be affected (the crew) would most likely not be in the vicinity of the impact point of the collision, where the damaged cask would be located.

The limiting accident is a ship collision severe enough to breach a cask carrying foreign research reactor spent nuclear fuel and also cause a large fire. Some of the radioactive contents of the cask could be released and carried into the air by the heated gases of the fire as a plume of radioactive particles. For an airborne release of this type to occur, the cask-carrying vessel must stay afloat during and immediately after the accident. In practice, this would mean that the ship must stay afloat for a period of some hours following an accident of the requisite severity. This latter condition must be satisfied for atmospheric dispersal to occur, even though marine casualty files indicate that a common outcome of severe ship collisions is rapid sinking, often within a matter of minutes. Assuming the cask was damaged by a severe collision; and the ship remained afloat despite the severe collision; and the cask was engulfed in flames for a time sufficient to release a radioactive plume, there would likely be no human population on the ocean (excluding the crew) who could be affected.

It is possible that the ship could be in coastal waters (i.e., beyond the port's sea buoy) at the time of this severe collision. Except in port, a ship is seldom within 16 km (10 mi) of a population center, so the port accident public risk analysis in the next section covers public risk in this scenario. The ship's crew and people onboard other vessels that may come to provide assistance could be exposed to any released radioactive material. The number of people potentially exposed would be less than that used in the port accident analysis for populations near a port [less than 1.6 km (1 mi) from the port]. Additionally, accident frequencies at sea tend to be lower than in-port accident frequencies. Therefore, both the consequences and risks for an accident at sea are covered by the results of the port accident analysis.

Risks associated with this type of accident at sea are covered by the risks of the same type of accident in ports because humans in the vicinity of the accident at sea are much fewer in number than even the least populated port.

Sunken Cask

The second scenario of concern is that a foreign research reactor spent nuclear fuel cask or casks would be sunk. This could be the result of the ship sinking, of the casks being somehow swept overboard, or of a ground transport accident on a causeway. Submersion of an intact cask would not necessarily result in a release of its contents, as spent nuclear fuel casks are designed to withstand at least a 15 m (50 ft) immersion. It has been demonstrated that cask seals will remain intact at much greater depths (DOE, 1994m). Should a loaded foreign research reactor spent nuclear fuel cask (damaged or undamaged) sink anywhere in the U.S. coastal waters, it will be recovered regardless of depth. U.S. Coastal waters in this case refers to waters within the 12 mile territorial limit. Recovery would be accomplished, even in the deepest parts of U.S. coastal waters, such as in Puget Sound, which reaches 305 meters or 1,000 feet (Encyclopedia Americana, 1991). Elsewhere in the world, spent nuclear fuel casks can, and likely would, be recovered from water up to 200 m (660 ft) deep, which is beyond the range typical of coastal and port depths. Typically 200 m (660 ft) is considered the limit of the continental shelf. Recovery at depths greater than 200 m (660 ft) is possible but is more difficult.

If a sunken cask containing foreign research reactor spent nuclear fuel were recovered, the effect on the marine environment would be minimal, even if the recovery effort required up to 1 year to complete. The release to the ocean water of radioactive particles from the spent nuclear fuel requires that first the metallic spent fuel corrode, then the radioactive particles escape from the cask. Even if the cask were damaged, the most likely damage to a spent nuclear fuel cask, either from mechanical trauma or excessive depth, would be failure of the seal. Seal failure would allow seawater to enter the cask to begin the corrosion of the metallic spent nuclear fuel, but the flow of water through the cask to carry out the radioactive material would be minimal due to the small cross sectional area of the failed seal. The decay heat from the spent nuclear fuel is low, thereby providing no driving force to expel water out of the cask through the failed seal.

If a cask was not recovered, the radioactive constituents of spent nuclear fuel would be released slowly over time into the surrounding waters. Some of the radioactive material would be removed from the water by adhesion to suspended sediments. Assuming a cask were submerged on the deep ocean bottom and not recovered, the peak human dose to an individual ingesting seafood harvested from the area in which the breached submerged spent nuclear fuel cask would be located would be 114 mrem per year. If a sunken cask in coastal waters was not recovered, the peak human dose is conservatively estimated to be 14,000 mrem per year. Consequences to humans and to marine biota are presented in Table 4-3. Other studies of similar circumstances indicate that the individual dose would be even lower (DOE, 1980). Uranium (the major constituent of the spent nuclear fuel) has been found not to bioaccumulate in fish, and bioaccumulates only slightly in crustaceans and mollusks (IAEA, 1976). The peak doses for humans, fish, crustaceans, and mollusks are presented in Table 4-3 in the situation where a chartered ship carrying eight casks might sink in deep ocean. Doses for humans and other animals are expressed in units of rem and rad, respectively. Rem is discussed in some detail in Section 4.1.3. While rem is only used for measuring human exposure to radiation, rad is used to measure exposure of nonhumans to radiation. Rad is a unit of absorbed dose from ionizing radiation.

The probability provided in Table 4-3 is the probability of one ship accident and loss of a cask during the entire program. The consequences are from one unrecovered cask. The program risk is the product of the probability and the consequences. Humans would not be the principally exposed species in a marine accident involving foreign research reactor spent nuclear fuel. Estimates were made of the dose to the biota received from a damaged cask containing foreign research reactor spent nuclear fuel. This analysis assumes that the cask would lay on the deep ocean floor where it would slowly release its radioactive inventory whether it was damaged in the collision or not.

Table 4-3 Impacts of Unrecovered Casks in Deep Ocean

	Probability	*Consequences*	*Program Risk*
MEI (human)	1.7×10^{-6}	114 mrem/yr	0.00019 mrem/yr
Fish	1.7×10^{-6}	640 rad/yr	1.1 mrad/yr
Crustaceans	1.7×10^{-6}	880 rad/yr	1.4 mrad/yr
Mollusks	1.7×10^{-6}	30,000 rad/yr	49 mrad/yr

Risks associated with the release of the contents of the spent nuclear fuel elements into the deep ocean are expected to be very small due to the low probabilities and limited consequences. The highest estimated risk to the MEI is 0.00019 mrem per year for every year that the cask leaks and this hypothetical individual ingests seafood harvested from near the cask. DOE and the Department of State assume that these conditions could apply for about 5 years, so the total MEI dose would be 0.00095 mrem. This translates into a maximum estimated MEI risk of 5×10^{-10} LCF. This means that this hypothetical individual's additional chance of incurring an LCF would be less than one in a billion. The risks to fish, crustaceans, and mollusks are low enough that no adverse impacts would be expected.

Probabilities, consequences, and risks were also calculated for the cases of unrecovered casks in coastal waters, both undamaged and damaged. The results are presented in Table 4-4, again in terms of rem for humans and rad for other animals. In coastal waters, cask recovery is considered likely (NEA, 1988), which makes the probabilities in Table 4-4 low. Comparing Tables 4-3 and 4-4 shows that the consequences of a sunken cask in coastal waters would be greater than in the deep ocean, but when multiplied by the probabilities, the risks are actually lower.

Table 4-4 Impacts of Unrecovered Casks in Coastal Waters

	Probability One Undamaged Cask	*Consequences One Undamaged Cask*	*Program Risk*
MEI (human)	2.3×10^{-8}	190 mrem/yr	4.3×10^{-6} mrem/yr
Fish	2.3×10^{-8}	77 mrad/yr	1.8×10^{-6} mrad/yr
Crustaceans	2.3×10^{-8}	81 mrad/yr	1.9×10^{-6} mrad/yr
Mollusks	2.3×10^{-8}	210 mrad/yr	4.8×10^{-6} mrad/yr
	Probability One Damaged Cask	*Consequences One Damaged Cask*	*Program Risk*
MEI (human)	4.6×10^{-11}	14,000 mrem/yr	6.4×10^{-7} mrem/yr
Fish	4.6×10^{-11}	620 mrad/yr	2.9×10^{-8} mrad/yr
Crustaceans	4.6×10^{-11}	660 mrad/yr	3.0×10^{-8} mrad/yr
Mollusks	4.6×10^{-11}	14,000 mrad/yr	6.4×10^{-7} mrad/yr

These risk estimates were derived assuming that the foreign research reactor spent nuclear fuel is shipped at a rate of one cask per voyage. Assuming a different shipping schedule, such as eight casks per voyage, would not result in a different estimate of the risks. The potentially higher consequences of an accident involving more than one shipping cask would be balanced by the reduced probability of an accident due to the reduced number of shipments. For example, the risk associated with one shipment of eight casks is equivalent to the risks associated with eight single cask shipments.

4.2.1.4 Marine Transport Cumulative Impacts

The cumulative impact of radioactive material shipments on ships' crews beyond that discussed in Section 4.2.1.2 was not estimated. In estimating the cumulative impact on port workers (see the following section) it was possible to estimate the total number of shipments of radioactive material through a port. However, it is not as simple to estimate the total number of shipments of radioactive material that involve the same ship and crew. It is expected that each ship's crew would be exposed to fewer of the shipments of radioactive material than that assumed for the port worker in the cumulative impact analysis for the port. For port workers, the impacts of the shipments other than the foreign research reactor spent nuclear fuel were of the same order of magnitude, but lower than the foreign research reactor spent nuclear fuel shipments. Therefore, the individual crew member's exposure from shipments other than the foreign research reactor spent nuclear fuel shipments would be a small fraction of the dose received due to the foreign research reactor spent nuclear fuel shipments.

4.2.1.5 Marine Transport Mitigation Measures

The principal environmental impact that would occur during marine transport would be radiation dose to the ships' crews. Most of this dose occurs because crew members must visually inspect the cargo every day for safety reasons, and the inspections cannot be curtailed.

The magnitude of the estimated impacts from this portion of the basic implementation of Management Alternative 1 is primarily due to two items: the conservative assumption that the radiation field emanating from all of the casks would be at the regulatory limit (as opposed to the levels of one-tenth of the regulatory limit that have been observed in past foreign research reactor spent nuclear fuel shipments), and the conservative assumption that the same crew member is involved in inspections for all of the casks on nine shipments during any given year. In reality, neither of these conservative assumptions would be likely to occur. Nevertheless, to ensure that no member of a ship's crew could receive a dose above what is allowed for a member of the general public, DOE would mitigate this effect by implementing a system through its shipping contractor to track each ship and crew involved in the shipment of foreign research reactor spent nuclear fuel. DOE would also include a clause in the contract for shipment of the foreign research reactor spent nuclear fuel requiring that other crew members be used if any crew member approaches a 100 mrem dose in any year.

If a cask or casks were sunk in deep ocean or coastal waters, DOE and the Department of State would employ modern underwater search techniques to locate and recover the cask(s), thus minimizing the potential impacts to marine life.

4.2.2 Port Activities Impacts

4.2.2.1 General Assumptions and Analytic Approach

To assess the range of potential impacts on ports at which a ship carrying foreign research reactor spent nuclear fuel might call, 13 ports of entry representing a wide range of port city population densities were selected for detailed evaluation. Eight of the ports—Charleston, SC; Elizabeth, NJ (for the New York City area); Philadelphia, PA; Norfolk, VA (representing Hampton Roads); Jacksonville, FL; Savannah, GA; Wilmington, NC; and Military Ocean Terminal at Sunny Point (MOTSU), NC—are East Coast ports that represent high, medium, and low population density ports. The Norfolk Terminal was selected to represent the three terminals (Newport News, Norfolk, and Portsmouth) at Hampton Roads for the analysis of potential impacts because this terminal provides the most conservative values in terms of estimated impacts. The West Coast ports chosen for evaluation were Long Beach, CA; Concord Naval Weapons Station (NWS), CA; Portland, OR; and Tacoma, WA, to represent high and medium population density ports. On the Gulf Coast, Galveston, TX was analyzed. These ports were selected to represent a range of ports in this analysis, not necessarily as the chosen ports of entry for foreign research reactor spent nuclear fuel. Ports representative of a group of ports with similar characteristics (i.e., of similar population around the port) were selected for analysis rather than attempting to analyze accidents at every potential port. Actual port selection and specific selection criteria are discussed in Appendix D, Section D.1.

The analysis assumed that there were no restrictions on the shipping routes taken by the cargo vessel carrying the foreign research reactor spent nuclear fuel. This assumption allows the vessel to make intermediate stops at any U.S. port capable of unloading the vessel. This implies that the vessel could enter most ports capable of receiving ocean-going cargo vessels, a group of ports that far outnumbers the ports that survive the port selection criteria for the receipt of foreign research reactor spent nuclear fuel. It was conservatively assumed that regularly scheduled commercial ships carrying foreign research reactor spent nuclear fuel would pass through two intermediate U.S. ports before reaching the port of entry for the foreign research reactor spent nuclear fuel. The 13 ports with high, medium, and low population densities that were chosen for site-specific accident analysis provide a perspective on the accident risks at the more than 100 ports that could be intermediate ports of call for the foreign research reactor spent nuclear fuel vessels.

Each port stop would or could involve:

- Port entry from the sea buoy,
- Docking,
- Inspection of cargo,

- Partial unloading of cargo,
- Partial reloading of cargo, and
- Port exit to the sea buoy.

As with the marine transport, the port impacts were evaluated for two conditions: incident-free and accident conditions. Summary results are presented in the following sections. Details of the analysis are presented in Appendix D.

4.2.2.2 Conservative Assumptions and Maximum Estimated Impacts of Incident-Free Port Activities

As stated in Section 2.6, no spent nuclear fuel transportation cask has ever released its contents (radioactive material), even as a result of an accident. For this reason, release of radioactive material is not considered as part of the incident-free analysis. The only impact considered is that caused by radiation exposure due to radiation emitted by foreign research reactor spent nuclear fuel contained within the transportation casks. Since no radioactive material would be released, there would be no impacts on land, water, or air quality in any of the ports or any of the waterways used by ships in the transport of foreign research reactor spent nuclear fuel.

Risks associated with the foreign research reactor spent nuclear fuel in incident-free conditions in port are predominantly those to inspectors and port workers. Port workers and inspectors are not radiation workers as defined by NRC regulations. Thus, the maximum allowable annual exposure for these personnel would be 100 mrem, the same radiation dose limit established by the NRC to protect individual members of the public (DOE, 1990c). When a ship arrives in its first port, the spent nuclear fuel package would be inspected by customs officials, U.S. Coast Guard personnel, and others. Up to six inspections, estimated at up to 15 minutes per person per spent nuclear fuel cask, were conservatively assumed. Once inspections are complete, the ship would partially unload and reload cargo. After that, DOE and the Department of State conservatively assumed that the ship would proceed to another intermediate port and then to the port of entry for the foreign research reactor spent nuclear fuel.

To determine the incident-free risks associated with port operations, two types of ships were considered for the shipment of the foreign research reactor spent nuclear fuel. In the first case, DOE and the Department of State conservatively assumed that all shipments were made on regularly scheduled commercial breakbulk ships. This type of vessel was selected because it maximized the time required for port activities, such as off-loading and inspections. In addition, during the operations at the intermediate port stops, DOE and the Department of State conservatively assumed that other unloading and loading operations would occur in the vicinity of the container with the loaded foreign research reactor spent nuclear fuel cask in one of the intermediate ports. Risks associated with these activities, which are comparable to the risks associated with the off-loading of the foreign research reactor spent nuclear fuel, have been included in the assessment. Transport of the material on this type of vessel would therefore result in the highest worker radiation doses in the incident-free analysis. All worker exposures were calculated by estimating the times required for activities and the distances from the transportation cask to where the worker would most likely be located.

To provide a measure of the difference in the worker exposures resulting from the use of cargo vessels other than the regularly scheduled commercial breakbulk vessels, the analysis was also performed for port operations associated with the use of a chartered container vessel. This type of vessel requires the least amount of time to unload. DOE and the Department of State also assumed that a chartered vessel would not make any intermediate port stops, so that the ship's port of entry into the United States would also be the port of entry for the foreign research reactor spent nuclear fuel. Use of these two types of vessels in the analysis provides an estimate of the range of the maximum incident-free risk associated with port operations.

At the port of entry, the casks would be off-loaded by port workers, and arrangements would be made for the immediate departure of the foreign research reactor spent nuclear fuel from the port. In recognition of instances where some delay may occur, DOE and the Department of State conservatively assumed a delay of up to 24 hours in a secure staging area. The 24-hour period for the staging of spent nuclear fuel casks was selected because it is possible that, on occasion, the spent nuclear fuel casks would not leave the secure staging area the same day that they arrived, depending on variables such as the time of day the casks clear customs and the weather. Nonetheless, DOE and the Department of State consider it unlikely that the casks would remain in the staging area for longer than 24 hours.

To estimate the maximum individual exposure, the shipments were divided into East Coast and West Coast shipments, depending on the country of origin. Spent nuclear fuel shipments from Europe, Africa, the Middle East and parts of South America were designated as East Coast shipments, all others were designated as West Coast shipments. Under these assumptions, the East Coast port(s) would receive approximately 535 casks and the West Coast port(s) approximately 186 casks. DOE and the Department of State also conservatively assumed for this analysis that all the shipments would pass through the same intermediate ports as the shipments on regularly scheduled commercial vessels and have the same port of entry.

Further, DOE and the Department of State made the very conservative assumption that the same inspectors and workers would handle every cask shipment. The per-shipment doses were then multiplied by the number of shipments for the East Coast to determine the maximally exposed worker dose for the basic implementation of Management Alternative 1.

In determining the worker population exposure, all shipments (East Coast and West Coast) were considered. This results in the integrated dose for the entire basic implementation of Management Alternative 1 which would span 13 years. The maximum estimated incident-free risks to port personnel due to the basic implementation of Management Alternative 1 are presented in Table 4-5. The incident-free risk to the general public would be zero because only workers would be near the casks in port.

This table shows the maximally exposed worker dose, worker population dose, and associated risks for the shipment of foreign research reactor spent nuclear fuel as containerized cargo on a regularly scheduled commercial breakbulk vessel and as cargo on a chartered container vessel. These figures represent the range of maximum estimated impacts for the various shipping modes available for the ocean transport of foreign research reactor spent nuclear fuel.

As the table shows, the highest estimated maximally exposed worker risk is 0.00052 LCF, which is based on the annual regulatory limit every year for 13 years. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

The highest total population risk for port workers is 0.012 LCF, which is much less than one LCF.

Table 4-5 Incident-Free Port Activity Impacts[a,b]

Impacts per Cask Transfer

Risk Group	Regularly Scheduled Commercial Breakbulk Ship: Maximally Exposed Worker Dose (mrem)	Regularly Scheduled Commercial Breakbulk Ship: Maximally Exposed Worker Risk (LCF)	Regularly Scheduled Commercial Breakbulk Ship: Population Dose to Workers per Cask (person-rem)	Regularly Scheduled Commercial Breakbulk Ship: Population Risk (LCF)	Chartered Container Ship: Maximally Exposed Worker Dose (mrem)	Chartered Container Ship: Maximally Exposed Worker Risk (LCF)	Chartered Container Ship: Population Dose to Workers per Cask (person-rem)	Chartered Container Ship: Population Risk (LCF)
Inspectors	3.8	0.0000015	0.013	0.0000052	1.3	5×10^{-7}	0.0053	0.0000021
Port Handlers, Intermediate Ports	2.2	9×10^{-7}	0.018	0.0000071	----	----	----	----
Port Handlers, Port of Destination	2.0	8×10^{-7}	0.0066	0.0000026	0.46	1.8×10^{-7}	0.0015	6×10^{-7}
Port Staging Personnel	0.36	1.4×10^{-7}	0.0045	0.0000018	0.4	2×10^{-7}	0.0046	0.0000018
Maximum	3.8	0.0000015	----	----	1.3	5×10^{-7}	----	----
Total	----	----	0.042	0.000017	----	----	0.011	0.0000045

Impacts for the Entire Basic Implementation

Risk Group	Regularly Scheduled Commercial Breakbulk Ship: Maximally Exposed Worker Dose (mrem)	Regularly Scheduled Commercial Breakbulk Ship: Maximally Exposed Worker Risk (LCF)	Regularly Scheduled Commercial Breakbulk Ship: Population Dose to Workers (person-rem)	Regularly Scheduled Commercial Breakbulk Ship: Population Risk (LCF)	Chartered Container Ship: Maximally Exposed Worker Dose (mrem)	Chartered Container Ship: Maximally Exposed Worker Risk (LCF)	Chartered Container Ship: Population Dose to Workers (person-rem)	Chartered Container Ship: Population Risk (LCF)
Inspectors	1,300[c]	0.00052[c]	9.4	0.0038	670	0.00027	3.8	0.0015
Port Handlers, Intermediate Ports	1,186	0.00047	13	0.0052	----	----	----	----
Port Handlers, Port of Destination	1,072	0.00043	4.8	0.0019	250	0.0001	1.1	0.00044
Port Staging Personnel	190	0.000076	3.2	0.0013	210	0.000084	3.3	0.0013
Maximum	1,300[c]	0.00052	----	----	670	0.00027	----	----
Total	----	----	30	0.012	----	----	8.2	0.0032

[a] *These results are based on the assumption that the dose rates associated with the casks are all based on the regulatory limit. Historically, the average of these dose rates has been equal to about one-tenth of the regulatory limit, so this assumption is conservative.*

[b] *These results are all based on the assumption that each voyage carries two casks. This assumption is conservative because chartered ships may carry up to eight casks.*

[c] *With all the conservative assumptions in this analysis, the maximally exposed worker dose could theoretically exceed the annual regulatory limit. Therefore, DOE would require mitigation measures to keep the maximally exposed worker dose down to 100 mrem per year or lower. These results are based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years. See Appendix D for maximally exposed worker doses without mitigation measures.*

4.2.2.3 Conservative Assumptions and Maximum Estimated Impacts of Accidents During Port Activities

Section 4.2.1.3 discussed the impacts of marine accidents that could occur either in the open ocean or during coastal passages. This section discusses the impacts of accidents that could occur anywhere from the sea buoy into the port and at the pier.

Methodology

An analysis of reasonably foreseeable accidents must evaluate the consequences of possible accidents and the probability of an accident occurring. In incident-free marine transport, some exposure would be expected from radiation emitted from the casks. In the case of accidents, the probability of exposure is only an estimate of a hypothetical event. Accident probabilities are derived from published maritime accident rates. The analysis of ship collisions concludes that only one hold of the ship carrying the foreign research reactor spent nuclear fuel transportation casks would be subject to sufficient forces to potentially result in cask damage. There is no difference between the risks associated with a single shipment with two casks in a hold, and two shipments of a single cask each. The consequences of the accident with two casks in the hold may be as large as twice the consequences of an accident involving one cask. But the probability of an accident involving the ship carrying the two casks is half the probability of one of the two ships carrying a single cask being involved in an accident. Therefore, the potential risk from accidents, marine transportation of spent nuclear fuel was modeled in the port accident analysis as occurring in one cask per shipment.

Because accidents can be of any degree of severity, from a "fender bender" to one involving severe impact and prolonged fire, the severity spectrum is divided into a number of accident severity categories. Each category is assigned a conditional probability of occurrence [i.e., a probability (given that an accident occurs) that it will be of that particular severity]. In general, the more severe the accident, the more remote the chance of such an accident. In this analysis, the accident severity spectrum is divided into six categories (Wilmot et al., 1981), which are discussed further in Appendix D. The accident scenarios considered in this analysis fall into the three most severe of the six severity categories.

Accidents in the first three, least severe, categories result in no release of material from the spent nuclear fuel transportation cask. These categories include all the accident scenarios associated with handling the spent nuclear fuel cask, including dropping the cask during transfer from the vessel to the truck or train. The transportation casks are certified to maintain their integrity when dropped from 15 m (50 ft) onto a perfectly unyielding surface. During the cask transfer, however, the crane may lift the cask higher than 15 m (50 ft). As the dock surface is softer than "perfectly unyielding," the soft surface of the dock would compensate for the greater drop height. Studies (DOE, 1994m) have shown that a cask can be dropped from much higher than the certification test height onto a yielding surface, without breaching.

The accidents analyzed in the three highest severity categories include collision of vessels, either in the approach to the harbor or when the vessel transporting the foreign research reactor spent nuclear fuel would be docked. The category 4 accident severity category models a collision of two vessels resulting in the breach of the transportation cask. Severity categories 5 and 6 model collisions that would breach the cask and subsequent fires that would cause the release of additional material, with category 6 fires being more intense than those for category 5.

As mentioned above, the spectrum of accidents, including a container breach and fire, were evaluated at two locations in each of the 13 ports of entry selected to envelop the port impacts. The approach to each port, from the sea buoy to the selected dock, was examined to determine the location where the accident would be most likely as well as have the largest consequence. This point is typically near the highest population center along the approach to the pier, and DOE and the Department of State conservatively selected this point for accident analysis. The second location where the spectrum of accidents was assumed to occur is at pier-side.

At these two locations, the probability of an accident was assigned, based on historical ship accident data (see Appendix D for details). These data were used to develop accident frequencies for collisions between vessels large enough to generate the forces sufficient to damage the cask (additional details on the development of the model used are provided in Appendix D), and to develop the frequency of collisions concurrent with fires (Lloyds, 1991). These data include information on a large number of ship voyages and accidents due to all causes. The cause of the accident (human error, weather, mechanical failure, etc.) was not identified for this analysis. However, the data apply to damage to or loss of a vessel and would include information on accidents that were caused by severe weather. Although severe weather accident scenarios are not specifically identified in the analysis, they are considered through the use of these data.

The consequence modeling for the port accident analysis was performed using the MELCOR Accident Consequences Code System (MACCS) (Jow, 1990), a code developed for the conservative modeling of accident consequences for nuclear powerplants and approved by the NRC. This code uses site-specific information, including population and meteorology, along with the identified radionuclide inventory and release fractions to determine the consequences of the accident scenarios. In determining the effects of the release of radioactive material, the MACCS code evaluates the direct dose to the public as well as several additional pathways including inhalation, ingestion, and groundshine. Groundshine is the dose received from radioactive material deposited on the ground's surface.

A conservative assumption incorporated into the risk assessment is that the entire population would remain in the area for 24 hours and therefore would be exposed to the greatest extent possible to radioactive material deposited on the ground from the plume. In reality, individuals close to an accident could be evacuated.

Atmospheric dispersion is usually the primary mechanism for dispersing any material that might be released in a severe accident. For the ship-collision-without-fire scenario (category 4), the release is modeled as occurring at the water surface level. For shipboard fires (categories 5 and 6), an elevated release due to the lifting effect of the fire is modeled. Meteorology data from the nearest National Weather Service Station were obtained for the 13 ports of entry selected as representative ports and input to the analysis of dispersion to ensure validity.

Cask Characteristics

The behavior of the cask in accidents within each accident severity category is accounted for in this analysis. "Type B" spent nuclear fuel casks (the kind in which the foreign research reactor spent nuclear fuel would be shipped) are massive, highly damage-resistant packages. Moreover, the spent nuclear fuel itself consists mostly of solid metallic materials that are not readily dispersed. Therefore, large releases would not be likely to occur, even in the most severe of accident conditions.

"Type B" packages are required to pass a series of rigorous tests that are associated with hypothetical accident conditions that might be encountered. These certification tests were developed by the International Atomic Energy Agency and promulgated as model regulations (IAEA, 1990). These model regulations have been adopted by the United States as well as all of the nations currently proposing to ship foreign research reactor spent nuclear fuel to the United States under the basic implementation of Management Alternative 1.

Ports Selected for Accident Analyses

Analyses of the impacts of possible accidents at representative ports were conducted. Thirteen ports were selected as being representative of the full range of ports in the United States, based on population and geography. Three of the ports are high-population density ports, two on the East Coast (Elizabeth, NJ and Philadelphia, PA) and one on the West Coast (Long Beach, CA). Five of the ports (Portland, OR; Tacoma, WA; Concord NWS, CA; Jacksonville, FL; and Norfolk, VA) are medium-population density ports, three on the West Coast and two on the East Coast. The remaining ports (MOTSU, NC; Galveston, TX; Savannah, GA; Wilmington, NC; and Charleston, SC) are low-population density ports. The 13 potential ports of entry for which accidents were analyzed collectively have a range of populations and geography that ensure that the results of these analyses are representative of the results that would have been reached if the analyses had been conducted for all ports. Additionally, these 13 ports include all 10 of the ports that meet all of the port selection criteria.

To demonstrate the representative nature of the analyses performed, a plot was made of the analyzed accident consequences for mean meteorological conditions at each port versus the port's population in a 16-km (10-mi) radius (Figure 4-1). The analyzed data points are shown as dots. The straight line represents the linear least squares fit of the data. Since the straight line represents an average of the data, some deviation from the line for individual data points is expected. The data fit well, with a correlation factor of 0.994455 (a correlation factor of 1.0 implies a perfect fit). This plot demonstrates the expected increase in the total population dose with an increase in port population. Deviations from the line by the calculated data are typically due to the distribution of population in relation to the local meteorology. Where most of the population is downwind of the port in normal weather, the corresponding population dose would likely be above the average line. For comparison, the total population dose due to background radiation is shown in the upper right corner. This comparison shows that population dose resulting from a severe accident would be approximately 0.2 percent of the annual background population dose.

As a check that the data from the 16-km (10-mi) radius population is valid, a similar analysis was performed correlating the 80-km (50-mi) radius population and accident consequences for seven ports. This analysis confirmed that the population dose as a function of population is linear, and therefore confirms that the range of ports selected for accident analysis fully covers the entire range of U.S. ports. More specific discussion of the results of the analyses is provided in Appendix D. This linearity of consequences and population show that any port selected for use as an intermediate port or port of entry for the foreign research reactor spent nuclear fuel, ranging from the least populous port (MOTSU) to the most populous port (Elizabeth) and including major ports of intermediate population, has had representative accident analyses performed.

Probabilities of Port Accidents

The probability of an accident occurring can be determined from historical statistics on ship collisions and mishaps. Maritime accident rate data from a Lloyd's of London database covering approximately 900,000 port calls by commercial vessels over a 15-year period (1978 to 1993) were examined to develop accident probabilities. The data indicate that the basic accident rate in and near ports is slightly less than 0.0001 accidents per port transit, or approximately 1 accident per 10,000 port visits.

Only the most severe accidents, however, would cause any damage to the cask. Thus, the conditional probabilities of occurrence of each accident severity were also developed from this database. As discussed in Appendix D, a conditional probability is defined as the probability, given that an accident has occurred, that it will be of a certain severity. To calculate overall probability of an accident of a particular severity, the base accident probability (accident rate) must be multiplied by the conditional probability.

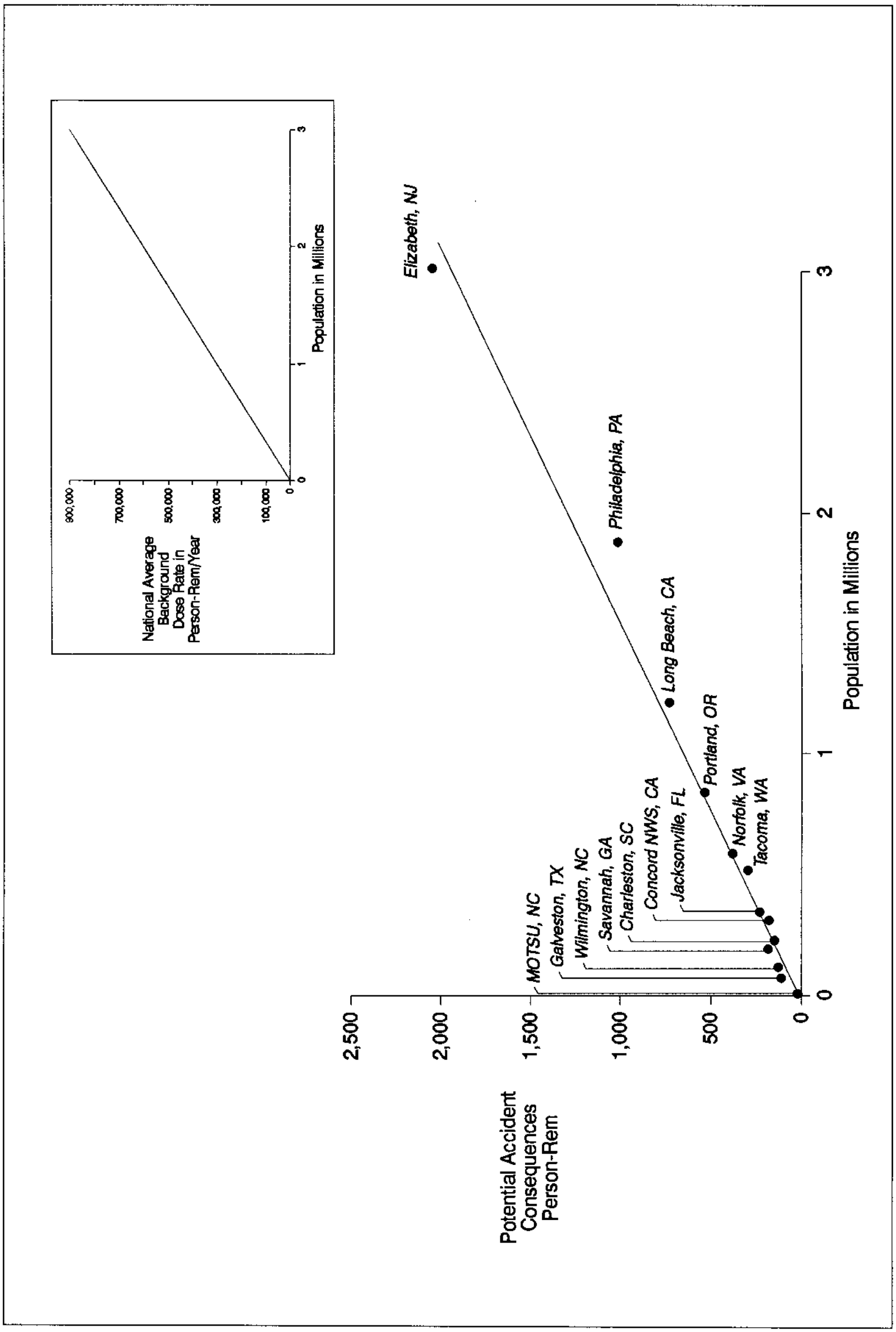


Figure 4-1 Consequences Versus Population [for a 16-km (10-mi) Radius]

Accidents are ranked according to their release categories. Release category 4 would result from the cask being damaged and compromised. Release category 5 would result from a damaged and compromised cask being enveloped in a fire. Release category 6 would result from a damaged and compromised cask being enveloped in a longer fire than a category 5 fire. The probabilities for the category 4, 5, and 6 accidents are 0.000006, 5×10^{-9}, and 6×10^{-10}, respectively. DOE and the Department of State assumed that it was equally likely that the accident occurs at the dock or in the channel, during the approach to the dock.

Consequences of Port Accidents

The consequence of an accident indicates the result, given that the accident were to occur, without any consideration for the likelihood of the accident occurring. The analysis conducted to determine the impacts of an accident involving foreign research reactor spent nuclear fuel in ports yields two different measures of the consequences. One measure is a calculation of the number of LCF that might result if the accident were to occur. These results are presented in Table 4-6 for the three most severe types of accidents under mean meteorological conditions.

The results presented in Table 4-6 are based on the mean consequences, so they are equivalent to results expected for the accidents in the respective release categories. These results are also based on the conservative assumption that accidents involve a cask carrying the highest inventory of nuclear material expected. Appendix D provides information on the consequences associated with the range of spent nuclear fuel types considered.

Examination of Table 4-6 shows that the most adverse consequence (2.9 LCF) arises from a Release category 5 accident in the channel approaching the Port of Elizabeth. This places the vessel just west (and generally upwind) of New York City. Although some of the release fractions change between categories 5 and 6, most of them do not. Therefore, the total population dose and the related number of LCF are about the same for Release categories 5 and 6. Release category 4 would be a release with no fire. In the absence of a fire, the release would remain at ground or water level without wide dispersion, hence the greatly reduced number of affected individuals and reduced consequences.

In addition to calculating the health effects of an accident on man, the MACCS code also calculates the impact of the accident on the land and structures around the accident site. These effects are characterized by the costs of activities required to bring the land and structures back into a usable condition. These activities are characterized as (1) no remedial action required; (2) decontamination – the resources can be returned to use immediately after clean-up; (3) interdiction – the resources must be temporarily abandoned, for several years, prior to their return to use; and 4) condemnation – the resources are considered unusable for an extended period. In all of the consequence analyses performed for each of the accident sites, there are no costs calculated that are associated with decontamination, interdiction, or condemnation. This means that all of the land and structures would be immediately available for use. (The consequences calculated by MACCS for the immediate vicinity of the accident are based on average value for the area within 1.6 km (1 mi) of the accident. Even though the average consequences calculated by MACCS show no costs associated with accident clean-up, the area immediately around the ship carrying the foreign research reactor spent nuclear fuel (i.e., the dock area) may require some remedial activity).

A sensitivity analysis was performed to address the potential impact of shipboard fires with extremely high temperatures that could result in the foreign research reactor spent nuclear fuel attaining temperatures above the melting point of the aluminum-based fuel or the combustion temperature of the TRIGA fuel. This analysis shows that the maximum consequences of such a fire are a factor of 100 larger than those

Table 4-6 Port Accident Consequences (LCF)

Locations	Accident Severity Category[a] 4	5	6
Elizabeth at the Dock	0.00010	2.8	2.7
Elizabeth in the Channel	0.00016	2.9	2.8
Long Beach at the Dock	0.000093	2.0	2.0
Long Beach in the Channel	0.000035	1.8	1.9
Philadelphia at the Dock	0.000078	1.2	1.2
Philadelphia in the Channel	0.000037	1.2	1.2
Portland at the Dock	0.000034	0.52	0.53
Portland in the Channel	0.000023	0.50	0.51
Norfolk at the Dock	0.000024	0.38	0.37
Norfolk in the Channel	0.000013	0.30	0.30
Charleston at the Dock (Wando Terminal)	0.000011	0.19	0.19
Charleston at the Dock (NWS Charleston)	0.0000068	0.22	0.22
Charleston in the Channel	0.000017	0.19	0.19
Tacoma at the Dock	0.000024	0.75	0.80
Tacoma in the Channel	0.000017	0.63	0.66
Concord NWS at the Dock	0.000019	0.90	0.96
Concord NWS in the Channel	0.000041	1.40	1.50
Jacksonville at the Dock	0.000012	0.31	0.31
Jacksonville in the Channel	0.000011	0.24	0.25
Savannah at the Dock	0.000025	0.23	0.23
Savannah in the Channel	0.0000059	0.18	0.19
Wilmington at the Dock	0.000017	0.22	0.23
Wilmington in the Channel	0.0000042	0.098	0.10
Galveston at the Dock	0.000032	0.64	0.70
Galveston in the Channel	0.000014	0.63	0.69
MOTSU at the Dock	0.0000032	0.099	0.11
MOTSU in the Channel	0.0000042	0.098	0.10

[a] *These accident release categories are the three highest in severity.*

calculated for the base case (Appendix D, Section D.5.4.2.2, Table D-31). An extremely high temperature ship fire is highly unlikely (one in ten billion per shipment) and the risks are comparable to those calculated in the base case. This analysis also addressed the impact of an accident on the land around the port. Using the same characterizations as described in the preceding paragraph, the largest mean impact distance is a decontamination and interdiction distance which is limited to approximately 300 m (1,000 ft). The analysis also shows that this distance is representative of the impact of this highly improbable type of accident at any of the ports included in the proposed action.

Risks

The calculated risk (probability times consequence) to the nearby population on a per-shipment basis assuming one cask per shipment and for the entire basic implementation of Management Alternative 1 is presented in Table 4-7. Each risk value is the sum of the risks from accident severity categories 4, 5, and 6. (A sensitivity study was performed to assess the risks associated with accidents that result in extremely high temperature fires. This sensitivity study was limited to an analysis of the per-shipment risks associated with shipment of spent nuclear fuel through the highest population density port, Elizabeth, NJ. Even though the consequences of this type of an accident are orders of magnitude larger than those calculated for the base case analysis, this type of event is highly improbable and the risks are comparable to those calculated in the base case. A more detailed comparison of the base case and sensitivity analyses is presented in Appendix D, Section 5.4.3.2.)

Table 4-7 Port Accident Risks

Port	Per Shipment of One Cask		Total All Shipments	
	Population Dose (person-rem)	Population Risk (LCF)	Population Dose (person-rem)	Population Risk (LCF)
Elizabeth via:				
• Two High Population Ports	0.00013	5.6×10^{-8}	0.070	0.000029
• One High and One Intermediate Population Port	0.00011	4.8×10^{-8}	0.060	0.000025
• One High and One Low Population Port	0.00011	4.5×10^{-8}	0.057	0.000024
• Two Intermediate Population Ports	0.000056	2.4×10^{-8}	0.030	0.000013
• One Intermediate and One Low Population Port	0.000051	2.2×10^{-8}	0.027	0.000011
• Two Low Population Ports	0.000046	2.0×10^{-8}	0.024	0.000010
• Direct	0.000042	1.8×10^{-8}	0.022	0.0000094
Long Beach via:				
• Two High Population Ports	0.00011	4.7×10^{-8}	0.058	0.000025
• One High and One Intermediate Population Port	0.000080	3.4×10^{-8}	0.042	0.000018
• One High and One Low Population Port	0.000071	3.0×10^{-8}	0.038	0.000016
• Two Intermediate Population Ports	0.000050	2.1×10^{-8}	0.026	0.000011
• One Intermediate and One Low Population Port	0.000041	1.8×10^{-8}	0.022	0.0000092
• Two Low Population Ports	0.000032	1.4×10^{-8}	0.017	0.0000072
• Direct	0.000028	1.2×10^{-8}	0.015	0.0000062
Philadelphia via:				
• Two High Population Ports	0.00011	4.5×10^{-8}	0.057	0.000024
• One High and One Intermediate Population Port	0.000088	3.7×10^{-8}	0.047	0.000020
• One High and One Low Population Port	0.000083	3.5×10^{-8}	0.044	0.000019
• Two Intermediate Population Ports	0.000031	1.4×10^{-8}	0.016	0.0000072
• One Intermediate and One Low Population Port	0.000026	1.1×10^{-8}	0.014	0.0000061
• Two Low Population Ports	0.000021	9.3×10^{-9}	0.011	0.0000049
• Direct	0.000017	7.5×10^{-9}	0.0092	0.0000040
Portland via:				
• Two High Population Ports	0.000090	3.8×10^{-8}	0.047	0.000020
• One High and One Intermediate Population Port	0.000059	2.5×10^{-8}	0.031	0.000013
• One High and One Low Population Port	0.000050	2.2×10^{-8}	0.027	0.000011
• Two Intermediate Population Ports	0.000029	1.3×10^{-8}	0.015	0.0000066
• One Intermediate and One Low Population Port	0.000020	9.0×10^{-9}	0.011	0.0000047
• Two Low Population Ports	0.000011	5.1×10^{-9}	0.0059	0.0000026
• Direct	0.0000073	3.2×10^{-9}	0.0039	0.0000017
Norfolk via:				
• Two High Population Ports	0.000095	4.0×10^{-8}	0.050	0.000021
• One High and One Intermediate Population Port	0.000076	3.2×10^{-8}	0.040	0.000017
• One High and One Low Population Port	0.000071	3.0×10^{-8}	0.037	0.000016
• Two Intermediate Population Ports	0.000019	8.3×10^{-9}	0.0098	0.0000044
• One Intermediate and One Low Population Port	0.000014	6.1×10^{-9}	0.0072	0.0000032
• Two Low Population Ports	0.0000088	4.0×10^{-9}	0.0046	0.0000021
• Direct	0.0000048	2.1×10^{-9}	0.0025	0.0000011
Charleston (Wando Terminal) via:				
• Two High Population Ports	0.000092	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000074	3.1×10^{-8}	0.039	0.000016
• One High and One Low Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000016	7.4×10^{-9}	0.0087	0.0000039

Port	Per Shipment of One Cask		Total All Shipments	
	Population Dose (person-rem)	Population Risk (LCF)	Population Dose (person-rem)	Population Risk (LCF)
• One Intermediate and One Low Population Port	0.000012	5.2×10^{-9}	0.0061	0.0000027
• Two Low Population Ports	0.0000066	3.1×10^{-9}	0.0035	0.0000016
• Direct	0.0000027	1.2×10^{-9}	0.0014	6.4×10^{-7}
Charleston (NWS Charleston) via:				
• Two High Population Ports	0.000093	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000074	3.1×10^{-8}	0.039	0.000017
• One High and One Low Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000017	7.5×10^{-9}	0.0084	0.0000039
• One Intermediate and One Low Population Port	0.000012	5.3×10^{-9}	0.0058	0.0000028
• Two Low Population Ports	0.0000068	3.2×10^{-9}	0.0032	0.0000016
• Direct	0.0000028	1.3×10^{-9}	0.0011	6.8×10^{-7}
Galveston via:				
• Two High Population Ports	0.000099	4.2×10^{-8}	0.052	0.000022
• One High and One Intermediate Population Port	0.000080	3.4×10^{-8}	0.042	0.000018
• One High and One Low Population Port	0.000075	3.2×10^{-8}	0.040	0.000017
• Two Intermediate Population Ports	0.000023	1.0×10^{-8}	0.012	0.0000053
• One Intermediate and One Low Population Port	0.000018	8.0×10^{-9}	0.0094	0.0000042
• Two Low Population Ports	0.000013	5.8×10^{-9}	0.0068	0.0000031
• Direct	0.0000090	4.0×10^{-9}	0.0047	0.0000021
Jacksonville via:				
• Two High Population Ports	0.000094	4.0×10^{-8}	0.050	0.000021
• One High and One Intermediate Population Port	0.000075	3.2×10^{-8}	0.040	0.000017
• One High and One Low Population Port	0.000070	2.9×10^{-8}	0.037	0.000016
• Two Intermediate Population Ports	0.000018	7.9×10^{-9}	0.0093	0.0000041
• One Intermediate and One Low Population Port	0.000013	5.7×10^{-9}	0.0067	0.0000030
• Two Low Population Ports	0.0000078	3.6×10^{-9}	0.0041	0.0000019
• Direct	0.0000038	1.7×10^{-9}	0.0020	9.0×10^{-7}
Savannah via:				
• Two High Population Ports	0.000093	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000074	3.1×10^{-8}	0.039	0.000016
• One High and One Low Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000017	7.5×10^{-9}	0.0088	0.0000039
• One Intermediate and One Low Population Port	0.000012	5.3×10^{-9}	0.0062	0.0000028
• Two Low Population Ports	0.0000068	3.2×10^{-9}	0.0036	0.0000017
• Direct	0.0000028	1.3×10^{-9}	0.0015	6.9×10^{-7}
Wilmington via:				
• Two High Population Ports	0.000092	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000073	3.1×10^{-8}	0.039	0.000016
• One High and One Low Population Port	0.000068	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000016	7.2×10^{-9}	0.0084	0.0000038
• One Intermediate and One Low Population Port	0.000011	5.0×10^{-9}	0.0058	0.0000026
• Two Low Population Ports	0.0000062	2.9×10^{-9}	0.0032	0.0000015
• Direct	0.0000022	1.0×10^{-9}	0.0012	5.3×10^{-7}
Tacoma via:				
• Two High Population Ports	0.000092	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000062	2.6×10^{-8}	0.033	0.000014
• One High and One Low Population Port	0.000053	2.3×10^{-8}	0.028	0.000012
• Two Intermediate Population Ports	0.000031	1.4×10^{-8}	0.017	0.0000072
• One Intermediate and One Low Population Port	0.000023	1.0×10^{-8}	0.012	0.0000053
• Two Low Population Ports	0.000014	6.1×10^{-9}	0.0072	0.0000032
• Direct	0.0000097	4.3×10^{-9}	0.0051	0.0000023

Port	Per Shipment of One Cask		Total All Shipments	
	Population Dose (person-rem)	Population Risk (LCF)	Population Dose (person-rem)	Population Risk (LCF)
Concord NWS via				
• Two High Population Ports	0.000099	4.2×10^{-8}	0.052	0.000022
• One High and One Intermediate Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• One High and One Low Population Port	0.000060	2.5×10^{-8}	0.032	0.000013
• Two Intermediate Population Ports	0.000038	1.7×10^{-8}	0.020	0.0000087
• One Intermediate and One Low Population Port	0.000029	1.3×10^{-8}	0.016	0.0000067
• Two Low Population Ports	0.000021	9.0×10^{-9}	0.011	0.0000047
• Direct	0.000017	7.1×10^{-9}	0.0088	0.0000038
MOTSU via:				
• Two High Population Ports	0.000091	3.9×10^{-8}	0.048	0.000020
• One High and One Intermediate Population Port	0.000072	3.1×10^{-8}	0.038	0.000016
• One High and One Low Population Port	0.000067	2.8×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000015	6.8×10^{-9}	0.0080	0.0000036
• One Intermediate and One Low Population Port	0.000010	4.6×10^{-9}	0.0054	0.0000024
• Two Low Population Ports	0.0000053	2.5×10^{-9}	0.0028	0.0000013
• Direct	0.0000013	6.2×10^{-10}	0.00069	3.2×10^{-7}

The column on the left indicates the port of entry for the foreign research reactor spent nuclear fuel and the possible combinations of two intermediate ports, as well as no intermediate ports (direct). The second and third columns present two measures of the risk on a per-shipment basis. These risks are based on the conservative assumption that all East and West Coast deliveries would follow the same route for all shipments. The fourth and fifth columns sum the per-shipment risks for all of the shipments, for the entire basic implementation of Management Alternative 1. The two columns under the heading of "Total All Shipments" are the product of the per-shipment risk data for each type of spent nuclear fuel cask with the number of casks of that spent nuclear fuel type. DOE and the Department of State conservatively assumed in these calculations that each port would receive all the casks.

Consider first the per-shipment population exposure risk for a shipment of foreign research reactor spent nuclear fuel to Elizabeth via two high-population density ports. This value, 0.00013 person-rem, is the risk from one cask shipment of the highest nuclear material inventory which would first pass through two high-population density ports, such as Boston and Philadelphia, then would be delivered to Elizabeth. The risk of this cask shipment would be the sum of the risks associated with each of the three ports, because an accident could occur in any of the ports. Since risk is the product of consequences and probability, and probability has no units, the risk would be expressed in the units of the consequences, in this case population exposure (person-rem).

Comparing the risk of sending the foreign research reactor spent nuclear fuel to Elizabeth via two high-population density ports (0.00013 person-rem) to sending the spent nuclear fuel directly to Elizabeth (0.000042 person-rem) shows that the risk would be cut by about two-thirds by eliminating the intermediate ports. This is expected, since the estimated overall risk is the sum of the risk at each of the three ports, and three high-population density ports would have roughly the same risks. Now compare the per-shipment risk of using a low-population density port, say MOTSU, via two low-population density ports. Table 4-7 indicates that this risk would be 0.0000053 person-rem, or about 25 times lower than the highest risk, Elizabeth via two high population ports. All per-shipment risks are conservatively based on the highest nuclear material inventory cask to maximize the potential risk.

The manner of evaluating the per-shipment risk of LCF in Table 4-7 is the same as for the per-shipment population exposure risk. Once again, shipping the foreign research reactor spent nuclear fuel through or into high-population density ports would increase the risk, as would using ships that pass through intermediate ports on their way to the port of entry.

The range of total population risks would be from 0.070 to 0.00069 person-rem for the population dose and from 0.000029 to 3.2 x 10^{-7} LCF for the risk, comparing shipping to Elizabeth via two high-population density ports and shipping to MOTSU without intermediate ports. The highest estimated population risk due to port accidents that might occur due to the basic implementation of Management Alternative 1 is 0.000029 LCF. This means that there would be less than a one in ten thousand chance of some member of the public incurring an LCF due to the basic implementation of Management Alternative 1 port transits.

The highest estimated MEI accident risk is conservatively determined by multiplying the accident probability by the consequences, in terms of dose to the MEI, of that accident. The MEI in this case is assumed to be an individual at the center of the plume less than 1.6 km (1 mi) from the accident. The highest average MEI doses calculated for the accident severity categories are: 0.11 mrem for category 4, 117 mrem for category 5, and 95 mrem for category 6. See Appendix D, Section D.5.4.2.2 for details. The reason MEI dose for category 6 is relatively lower than that for category 5 is because the larger category 6 associated fire would disperse the radioactive material faster and farther than the category 5 fire. For the 721 shipments in the basic implementation of Management Alternative 1, and using the per port transit accident probabilities in Appendix D, the highest MEI accident risk is estimated to be 0.00042 mrem. This corresponds to about 2 x 10^{-10} LCF. This means that the chance of the MEI incurring an LCF due to a port accident under the basic implementation of Management Alternative 1 would be less than one in a billion.

Emergency Management and Response

Emergency response capabilities for a foreign research reactor spent nuclear fuel mishap would be available through the U.S. Coast Guard and the local jurisdictions surrounding each candidate port of entry, with specialized support available from DOE. The specialized analysis and identification of potential hazards, use of the robust "Type B" packaging, specific emergency plan and procedure development, training, response rehearsal, and interagency coordination for efficient and effective response would minimize the potential consequences should a foreign research reactor spent nuclear fuel mishap occur. The specific emergency management and response capabilities and responsibilities are described in Chapter 2, Section 2.7.

At military ports, the U.S. Coast Guard routinely provides safety/security screen escorts. The addition of foreign research reactor spent nuclear fuel shipments would have almost no effect on their ongoing operations.

Consequences of Port Accidents

A sensitivity analysis was performed to address the potential impact of extremely high temperature fires, fires that could result in the foreign research reactor spent nuclear fuel attaining temperatures above the melting point of the aluminum based fuel or the combustion temperature of the TRIGA fuel, on the consequences of an accident in port. This analysis, which uses the Port of Elizabeth, NJ as the site of the accident, is presented in Appendix D, Section D.5.4.3.2, and shows that even though the consequences of this type of an accident are two orders of magnitude larger than those calculated for the base case analysis, this type of event is highly improbable and the risks are comparable to those calculated in the base case.

This analysis also addressed the impact of an accident on the land around the port. Using the same characterizations as described in the preceding paragraph, the largest mean impact distance is a decontamination and interdiction distance which is limited to approximately 300 meters (1000 feet). The analysis also shows that this distance is representative of the impact of this highly improbable type of accident at any of the ports included in the proposed action.

4.2.2.4 Cumulative Impacts of Port Activities

Port workers are expected to be exposed to other shipments of radioactive materials in addition to those associated with the basic implementation of Management Alternative 1. These shipments include DOE and commercially initiated programs. An assessment has been made of the cumulative impact of the incident-free dose to the maximally exposed worker from all of these activities. The cumulative analysis is based on data collected at several ports for 2.5 years (January 1992 through June 1994). The maximally exposed port worker is estimated to receive less than 10 mrem per year from commercial shipments. Details of this analysis are presented in Appendix D, Section D.4.6. As previously stated, based on cask dose rates equal to the regulatory limit, the maximally exposed port worker could receive an annual dose greater than the NRC and DOE regulatory limit of 100 mrem per year (NRC, 1991). Therefore, DOE would implement mitigation measures.

4.2.2.5 Port Activities Mitigation Measures

As with marine transport, the principal environmental impact that would occur during port activities is radiation dose to workers. No members of the general public would be close enough to the transportation cask to receive any radiation dose. The workers would receive this dose during safety inspections and handling activities which cannot be curtailed.

Two conservative assumptions in this analysis drive the maximally exposed worker dose higher than would actually be expected. The radiation dose rate near every foreign research reactor spent nuclear fuel shipping container is assumed to be equal to the regulatory limit and the same individual is assumed to conduct all the inspections. Neither of these is actually likely to occur.

Nevertheless, DOE and the Department of State would require, through a clause in the shipping contracts, some administrative controls on the port workers to mitigate the radiation doses to the workers during inspection and handling activities. DOE and the Department of State would implement a system to track the inspectors and other port workers actually involved in the shipment of foreign research reactor spent nuclear fuel. If any inspector's or worker's dose approaches 100 mrem in any year, then DOE and the Department of State would require other inspectors or workers to be used. In this way, the maximally exposed worker dose would be constrained to the regulatory limit.

If a cask or casks were sunk in coastal waters, DOE and the Department of State would employ modern underwater search techniques to locate and recover the cask(s), thus minimizing the potential impacts to marine life.

4.2.2.6 Environmental Justice at the Port(s)

Executive Order 12898 deals with the issue of environmental justice and directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse human health or environmental effects of their programs, policies, and activities on minority and low-income populations.

The concept of environmental justice is discussed in more detail in Appendix A. During normal port activities associated with receipt of the foreign research reactor spent nuclear fuel shipments—including harbor activities, unloading the ship, transfer of the spent nuclear fuel containers to truck or train, and movement out of the port city—the dominant radiological impacts have been shown to be the exposures received by the workers in the immediate vicinity of the shipping container. These individuals include the inspectors, shipping container handlers, truck drivers, etc. Since the intensity of the gamma radiation falls off rapidly with distance, the doses that might be received by other workers and members of the general population can in theory be calculated, but would not generally be measurable or distinguishable from natural background radiation.

Potential radiological impacts to people residing near the port are associated with low probability (less than one in a million) accidents that are so severe that the spent nuclear fuel casks would be ruptured and a fire would burn long enough around the cask that some of the radioactive material would be released. In this case, some of the radioactive spent nuclear fuel might be vaporized and lifted by the heat of the fire and carried downwind of the accident location. Where and how far this radioactive material would go before being deposited on the ground would depend on how high the heat from the fire lofts it and the particular weather conditions at the time. Most of this vaporized spent nuclear fuel would be expected to be deposited in the first few kilometers downwind of the fire but small amounts could be carried out for several tens of kilometers.

Because the particular details of both the accident conditions (such as the severity of a fire) and the weather conditions at the time of an accident could vary so much, a range of accident conditions and wind directions, wind speeds, and other weather conditions were examined during the evaluation of accidents (see Section 4.2.2.3). Population impact evaluations were performed for distances out to 80 km (50 mi). The risk of LCF was found to be so small that zero LCF would be expected due to accidents at ports.

Appendix A describes minority populations and low-income households residing near the ports. Calculations for incident-free and accident conditions clearly demonstrate that for the general population, including minority and low-income groups, the radiological impacts would be very low. Minority or low-income populations living near the potential ports of entry would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. They would be subject to the same very low impacts as would the general population.

Implementation of the proposed action would have extremely low nonradiological effects on the environment at candidate ports, including the social and economic status of the general population, minority populations, and the low-income population surrounding candidate ports. Economic benefits that would result from increased cargo handling and transportation in the port area would be extremely small for the general population or any particular segment of the population residing near candidate ports.

4.2.3 Ground Transport Impacts

Foreign research reactor spent nuclear fuel is transported in large, heavy containers called transportation casks. Transportation casks are designed and constructed to contain the radioactivity in spent nuclear fuel during severe transportation accidents. NRC has estimated that transportation casks will withstand 99.4 percent of truck and rail accidents without sustaining damage sufficient to breach the transportation cask (NRC, 1987). Only in the worst conceivable conditions, which are of low probability, could a transportation cask of the type used to transport spent nuclear fuel be so damaged that there is a reasonable possibility of release of radioactivity to the environment.

Spent nuclear fuel has been transported along highways, railways, and waterways since 1949. Federal standards describe the routing requirements for spent nuclear fuel shipments. Spent nuclear fuel transported includes foreign research reactor, commercial, naval, and DOE spent fuel. Since 1949, there have been 21 incidents involving vehicles carrying irradiated fuel elements. None of these incidents resulted in damage to the structural integrity of a cask or the release of the cask's contents.

4.2.3.1 Conservative Assumptions and Analytic Approach

Transportation impacts may be divided into two parts: the impacts due to incident-free transportation and the impacts due to transportation accidents. For incident-free transportation and transportation accidents, impacts may be further divided into two parts: nonradiological impacts and radiological impacts. The nonradiological impacts consist of the vehicular impacts of transportation, such as vehicular emissions and traffic accidents.

For incident-free transportation, the radiological impacts would result from the radiation field that surrounds the cask. For transportation accidents, the radiological impacts would be based on the radioactivity released from the spent nuclear fuel transportation cask during the accident. Impacts are estimated for workers and the population along the transportation route.

For both incident-free transportation and transportation accidents, methodology developed by NRC and used by DOE in the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (SNF&INEL Final EIS) (DOE, 1995c) was used to estimate the impacts for foreign research reactor spent nuclear fuel in this EIS. These impacts were quantified as the estimated number of radiation-related cancer fatalities and the estimated number of nonradiological fatalities from vehicular emissions and traffic accidents. Appendices B, C, D, E, and F of this EIS contain more details on the equipment, regulations, and experience associated with spent nuclear fuel transportation, and the methodology, data, and conservative assumptions used to develop these estimates.

Under the basic implementation of Management Alternative 1, acceptance of the foreign research reactor spent nuclear fuel would require the transport of approximately 837 casks from seaports and Canadian border crossings to DOE facilities. The number of casks was determined by assigning the spent nuclear fuel from each foreign research reactor to a reasonably available and capable cask. Conservative assumptions were used to estimate cask capacity, which is based on physical, thermal, and radiological characteristics of the spent nuclear fuel. Appendix B contains more details on the foreign research reactor spent nuclear fuel transportation casks.

For the purposes of analysis in this EIS, the initial ground transportation activities to the Savannah River Site and/or the Idaho National Engineering Laboratory is called Phase 1, and the possible subsequent intersite ground transport and continued management is called Phase 2. The impact assessment includes analysis of between 13 and 161 intersite shipments, depending upon the mode of transportation (truck or rail) and the potential foreign research reactor spent nuclear fuel management sites that might be selected. Intersite shipments would be fewer than foreign shipments because the spent nuclear fuel would be cooler. Larger casks would likely be used, and more foreign research reactor spent nuclear fuel would be consolidated.

The first step in the ground transportation analysis was to determine the incident-free and accident risk factors, on a per-shipment basis assuming one cask per shipment, for transportation of the various spent nuclear fuel casks. Risk factors, as any risk estimate, are the product of the probability of exposure and the magnitude of the exposure. Accident risk factors were calculated for radiological and nonradiological traffic accidents. The probabilities and the magnitudes of exposure are discussed in Appendix E. Incident-free risk factors were calculated for crew and public exposure to radiation emanating from the cask, and public exposure to the chemical toxicity of the transportation vehicle exhaust. The probability of incident-free exposure and the magnitudes of exposure are discussed in Appendix E.

Calculation of risk factors was accomplished by first using the HIGHWAY (Johnson, et al., 1993a) and INTERLINE (Johnson, et al., 1993b) computer codes to choose representative routes in accordance with the U.S. Department of Transportation regulations. These codes provide population estimates along the routes so that the RADTRAN (Neuhauser, 1993) and RISKIND (Yuan, et al., 1993) codes could be used to determine the risk factors associated with ground transportation activities. These computer codes are described in more detail in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and Appendix E of this EIS.

The single largest contributor to the ground transport population doses (about 80 percent) calculated with RADTRAN was found to be the dose to members of the public at truck stops. The parameters used to calculate doses during truck stops are quite conservative. The parameters are based on the assumption that stops occur as a function of distance, with a truck stop rate of 0.011 hr per km (0.018 hr per mi). This stop rate results in over an hour of stop time per 100 km (62 mi) of travel. It was further assumed that at each stop, an average of 50 people are exposed at a distance of 20 m (66 ft). These parameters were used because they are the default parameters in the RADTRAN code and they were used in the Programmatic SNF&INEL Final EIS (DOE, 1995c). These conservative assumptions that are built in the code are highly unlikely to occur.

The next step is to use the risk factors and the number of shipments to estimate the risk of every possible way the foreign research reactor spent nuclear fuel program could be implemented. Because of the large number of ports, cask types, spent nuclear fuel types, and implementation options, simplifying assumptions are needed to control the amount of repetitive analysis:

- A review of the accident risk factors for the various types of spent nuclear fuel (see Appendix E) indicates that there is relatively little variation between the different types of foreign research reactor spent nuclear fuel, thus, it is not overly conservative to use the highest risk factors for all shipments.

- Spent nuclear fuel from countries bordering the Atlantic Ocean and Mediterranean Sea was assumed to arrive on the East Coast of the United States. Spent nuclear fuel from countries bordering the Indian and Pacific Oceans was assumed to arrive on the West Coast. This is conservative from an overland transportation standpoint, because, as shown in Appendix E, marine shipment to the coast nearest the management site would reduce the risk factors for the overland shipment.

- To account for the return transport of empty casks, the impacts due to vehicle emissions and traffic accidents were multiplied by two.

The foreign research reactor spent nuclear fuel could arrive at any of the ports of entry selected by DOE and the Department of State using criteria that are detailed in Appendix D, and would be likely to arrive at a variety of these ports. Therefore, the proposed impacts were completely analyzed three times, consisting of an upper bounding case, a lower bounding case, and an average case, for both truck and rail shipments. The upper bound case conservatively assumes the port(s) with the highest risk factors was chosen for each transportation activity. The risk factors are generally a function of distance and total population along the port to management site route, so the port chosen often shifted between Phase 1 and Phase 2. Conversely, the lower-bound case assumes ports with the lowest risk factors.

The average case is designed to provide a realistic estimate of the ground transport risk of transporting the foreign research reactor spent nuclear fuel. The risk factors are an arithmetic average of the risk factors for all acceptable ports. This represents the risk associated with the basic implementation of Management Alternative 1 and receiving foreign research reactor spent nuclear fuel at a variety of commercial ports.

Since each potential port of entry and each management site is capable of receiving spent nuclear fuel via rail or highway, the program was analyzed using each mode of transportation. The exception to this is the Nevada Test Site which has no existing rail capability, so that link was approximated by a hypothetical rail line to the Yucca Mountain Site. Additionally, the potential to use trucks to carry the relatively small casks from ports to potential foreign research reactor spent nuclear fuel management sites and rail to carry larger casks between potential foreign research reactor spent nuclear fuel management sites was analyzed. Site to site shipment would not occur until approximately 2006, so it is difficult to precisely predict which cask would be used. The analysis is based on a truck cask that carries 4 times as much spent nuclear fuel as a foreign cask, and a rail cask that carries 10 times as much spent nuclear fuel.

4.2.3.2 Impacts of Incident-Free Ground Transport

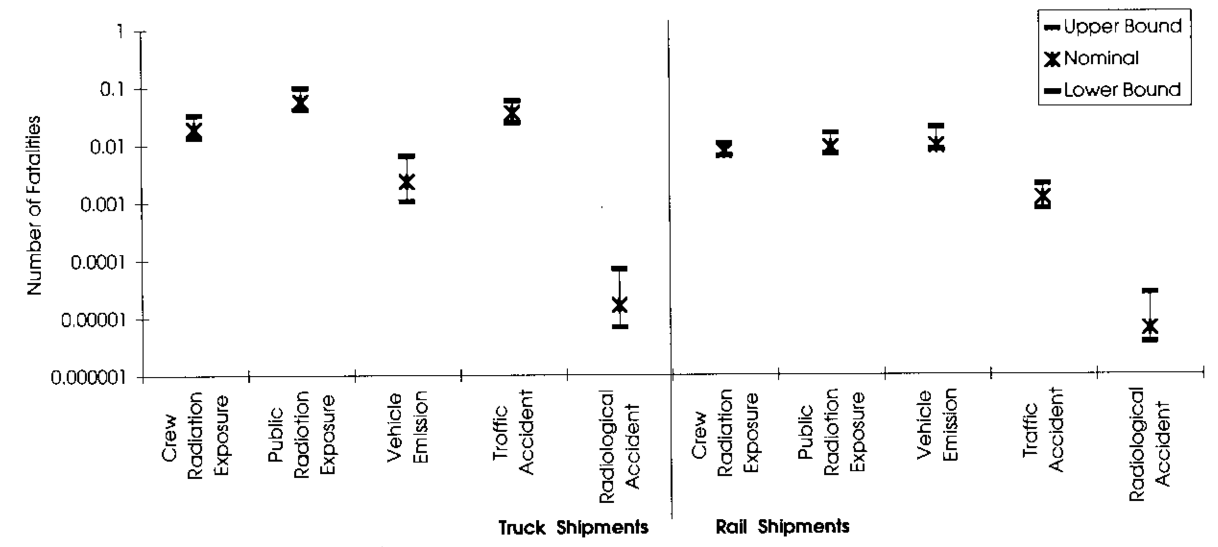
The incident-free transportation of spent nuclear fuel was estimated to result in total population risk that ranged from 0.013 to 0.30 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the transportation workers. Thus, the calculated maximum risk value for overland transportation is less than one fatality from cancer due to the basic implementation of Management Alternative 1. The range of fatality estimates is caused by two factors: (1) the option of using truck or rail to transport spent nuclear fuel; and (2) combinations of Phase 1 and Phase 2 sites that create varying cask shipment numbers and distances.

The estimated number of LCF due to radiation exposure for transportation workers ranged from 0.006 to 0.071. The estimated number of radiation-related LCF for the general population ranged from 0.007 to 0.22, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.001 to 0.052. These incident-free results apply to the workers and the public because both would be close enough to the cask to receive some radiation dose.

The impacts of transportation which are based on four Programmatic SNF&INEL Final EIS (DOE, 1995c) programmatic alternatives are summarized in Figures 4-2 through 4-5. The impacts of these additional programmatic alternatives are described in more detail in Appendix E.

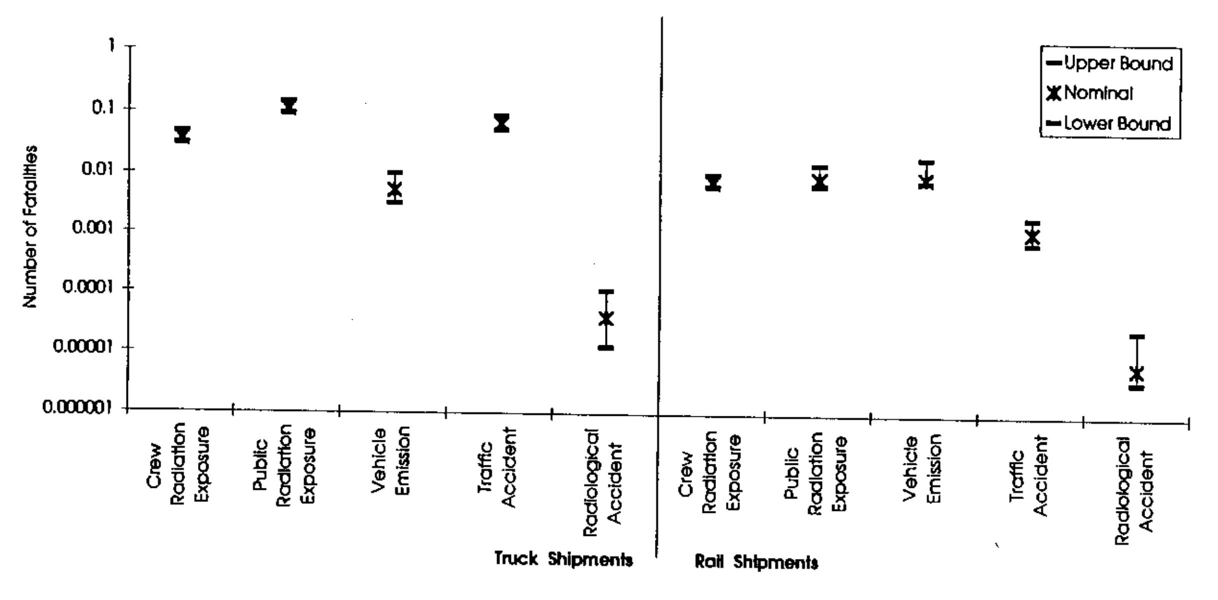
The highest estimated ground transport maximally exposed worker risk is 0.00052 LCF, just like the marine transport and port worker risks. This estimate is based on the conservative assumption that one truck driver makes enough trips to reach the regulatory limit of 100 mrem per year every year for 13 years. This means that under the assumptions described above, the chance of this individual incurring an LCF due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

The highest estimated incident-free population risk is 0.30 LCF, which means that there would be a 30 percent chance of one additional cancer fatality among the public and the ground transport workers due to the basic implementation of Management Alternative 1.



- *Analysis of Decentralization applies to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives*
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-2 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Decentralization Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-3 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative

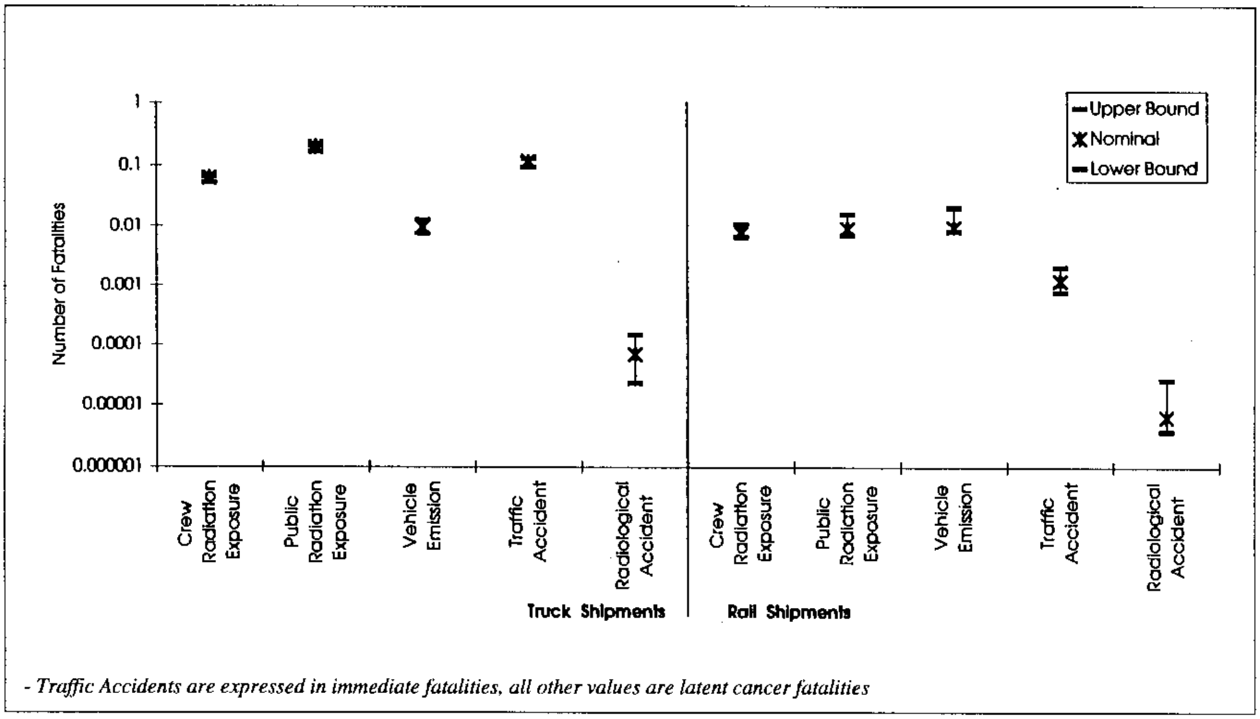


Figure 4-4 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative

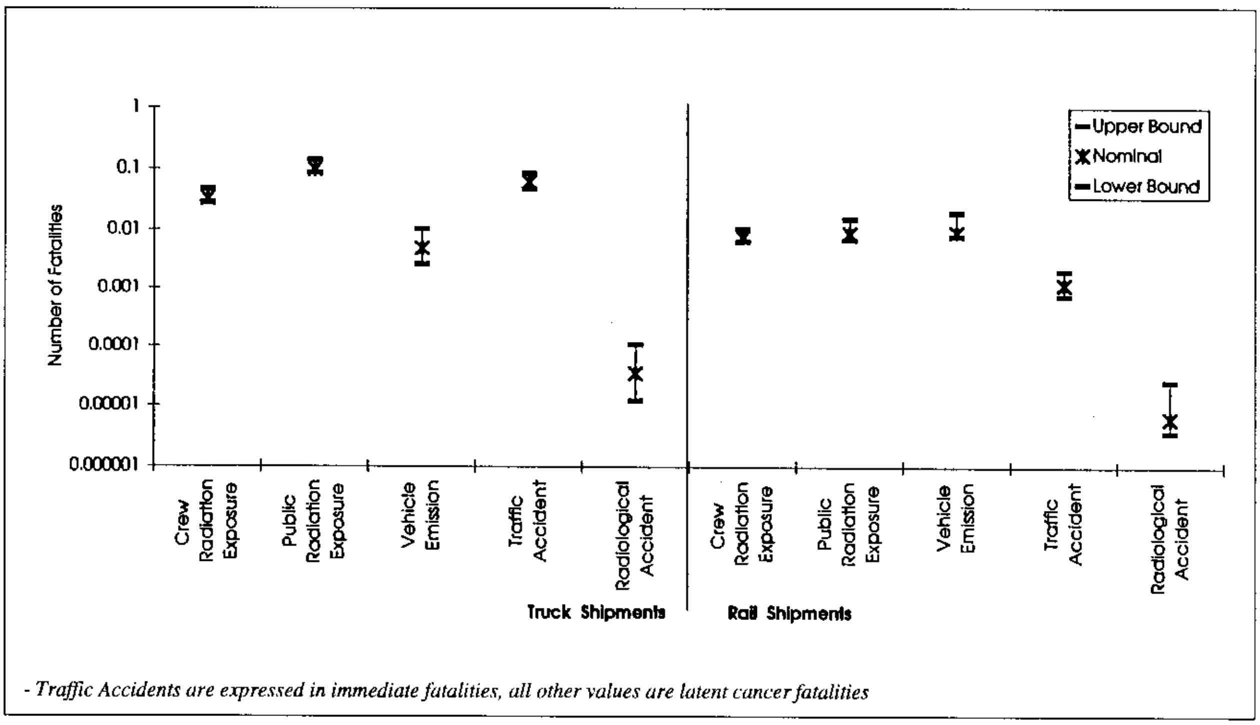


Figure 4-5 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

4.2.3.3 Impacts of Accidents During Ground Transport

The most severe accidents that might reasonably occur on this leg of the journey are truck or train crashes, followed by a large fire. If an accident occurred on a causeway at or near a port that caused a cask to fall into seawater, the consequences would be the same as if a cask fell off a ship into seawater. These consequences are presented in Section 4.2.1.3 under the subheading "Sunken Cask." Each State, and most local jurisdictions, maintain a hazardous materials response capability and a radiological protection program. These capabilities, along with the DOE radiological response assets that would be on-call for immediate technical assistance and response, would provide a high-level of expertise and would reduce the potential impacts of a foreign research reactor spent nuclear fuel accident.

Since hazardous materials team training is required to include radiological materials response, each team possesses a basic level of understanding and capability for a foreign research reactor spent nuclear fuel incident response. An incremental enhancement for spent nuclear fuel-specific response characteristics and planning may be required, especially for those jurisdictions along selected routes whose emergency responders are primarily volunteer organizations.

The development of a transportation plan specifically for the shipping campaign that would incorporate and integrate State and local emergency response plans, would increase emergency responder effectiveness and reduce the potential consequences of a foreign research reactor spent nuclear fuel accident.

Each State's emergency planning infrastructure, using the Local Emergency Planning Committees to the State Emergency Response Commission, enables these jurisdictions to identify and resolve potential emergency management and response issues and communicate issues that would require DOE and Department of State attention. This, along with DOE's Transportation External Coordination/Working Group, would ensure that all concerned agencies would be involved in the planning process to address potential problems before they become major hazards.

Risks

The total ground transportation accident risks for the basic implementation of Management Alternative 1 are estimated to range from 0.000004 to 0.00028 LCF from radiation and from 0.001 to 0.14 for traffic fatality, depending on the transportation mode and potential foreign research reactor spent nuclear fuel management sites that might be selected. Section 4.10 compares these risks to those of common activities. The reason for the range of fatality estimates is the same as those described for incident-free transportation. The risk of 0.14 for a traffic fatality means that under these conservative assumptions there would be a 14 percent chance of a traffic fatality related to the basic implementation of Management Alternative 1.

The maximum foreseeable offsite transportation accident would involve a shipment of foreign research reactor spent nuclear fuel in a suburban population zone under neutral (average) weather conditions. The accident has a probability of occurrence of about 0.0000001 per year (one chance in ten million), and could result in 14 person-rem and no fatalities. The probability of an accident occurring is at least an order of magnitude smaller in either an urban area or under stable atmospheric conditions. The consequences are less than an order of magnitude larger.

The impacts of transportation accidents are summarized in Figures 4-2 through 4-5, as described in the previous section, and are described in more detail in Appendix E. These tables can be used to assess the bounded absolute and relative risk under each representative Programmatic SNF&INEL Final EIS alternative.

The highest estimated MEI radiological risk to members of the public due to accidents during ground transport is 1.4 x 10^{-11} LCF. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in ten billion.

The highest estimated population radiological risk due to accidents is 0.00028 LCF, which is much less than one LCF.

4.2.3.4 Ground Transport Cumulative Impacts

The Programmatic SNF&INEL Final EIS (DOE, 1995c) analyzed the cumulative impacts of ground transportation, taking into account impacts from: (1) historical shipments of spent nuclear fuel to the five proposed foreign research reactor spent nuclear fuel management sites; (2) the programmatic alternatives; (3) other reasonably foreseeable actions that include transportation of radioactive material; and (4) general radioactive materials transportation that is not related to a particular action. The transportation of foreign research reactor spent nuclear fuel is included in the calculated totals under the spent nuclear fuel shipments for the Programmatic SNF&INEL Final EIS Alternatives 1 through 5. Proposed transportation of all spent nuclear fuel (of which the foreign research reactor fuel is a small component) accounts for less than one percent of the total LCF attributable to the transportation of radioactive material, and foreign research reactor spent nuclear fuel accounts for less than one quarter of that one percent. The total number of LCF over the time period 1943 through 2035 was estimated to be 290.

4.2.3.5 Ground Transport Mitigation Measures

The principal environmental impacts that would occur during ground transport are: (1) LCF due to radiation exposure, (2) LCF due to vehicular emissions, and (3) immediate fatalities due to traffic accidents. All three of these would be reduced by choosing port(s) of entry close to the management site(s). This would minimize the distance that must be covered by the vehicle(s).

Furthermore, in the case of truck transport, the truck driver(s) would be monitored for radiation dose. The annual maximally exposed worker limit of 100 mrem would never be approached during any single shipment, but the same driver could be used for multiple shipments throughout a year. DOE would implement mitigation measures through the foreign research reactor spent nuclear fuel acceptance contracts to ensure that each individual driver's dose remains below the regulatory limit. If any individual truck driver accumulates a dose approaching this limit in a year, DOE would require that new driver(s) be used to keep each individual driver's dose below the regulatory limit.

Since the casks would produce a radiation field of less than 10 mrem/hr at 2 m (6.6 ft) from the vehicle, an individual member of the general public would have to be within 2 m (6.6 ft) of the vehicle for at least ten hours in a year to receive a dose equal to the regulatory limit of 100 mrem/yr. A truck is not likely to sit in a traffic jam right beside another vehicle for as long as ten hours and an individual gas station attendant is not likely to spend ten hours refueling the trucks carrying foreign research reactor spent nuclear fuel. Therefore, DOE does not plan to implement ground transport mitigation measures for members of the general public.

4.2.3.6 Barge Transport

DOE and the Department of State have examined the possibility of using barges for the transport of foreign research reactor spent nuclear fuel as a substitute for truck or rail transport. The only two locations where barge transport is feasible are from the Port of Portland, OR up the Columbia River to the Hanford

Site and from the Port of Savannah, GA up the Savannah River to the Savannah River Site. Barge transport could only be implemented if one or both of these port/site combinations is selected in the Record of Decision.

For barge transport up the Columbia River, the incident-free radiological risk to the public would be approximately 0.0000043 LCF per cask shipment. This is slightly lower than the similar truck and rail shipment risks, which would be 0.000029 and 0.0000058 LCF per shipment, respectively. For barge transport up the Savannah River, the incident-free radiological risk to the public would be approximately 0.0000019 LCF per cask shipment. This is slightly lower than the similar truck and rail shipment risks, which would be 0.000028 and 0.0000026 LCF per shipment, respectively.

For barge transport up the Columbia River, the accident radiological risk due to both airborne and waterborne pathways would be approximately 3.5×10^{-8} LCF per cask shipment. This is slightly higher than the similar truck and rail shipment risks, which would be 1.5×10^{-8} and 3.8×10^{-9} LCF per shipment, respectively. For barge transport up the Savannah River, the accident radiological risk due to both airborne and waterborne pathways would be approximately 2.9×10^{-8} LCF per cask shipment. This is slightly higher than the similar truck and rail shipment risks, which would be 9.4×10^{-9} and 1.1×10^{-9} LCF per shipment, respectively.

The barge transport analysis is presented in more detail in Appendix E, Section E.8.15. The net result is that the foreign research reactor spent nuclear fuel could be transported by barge with approximately the same level of risk to workers and the public as if it was transported by truck or rail.

4.2.3.7 Environmental Justice Along Ground Transport Routes

The dominant radiological risks and impacts associated with incident-free transportation activities are the exposures received by the workers in the immediate vicinity of the casks and people who might be near the casks at truck stops. These individuals would be the only people receiving a measurable exposure during a spent nuclear fuel shipment. As discussed in Section 4.2.3.2, the number of radiation-related latent cancer deaths among transportation workers and the general public combined was calculated to be less than one. The same is true for cancer due to vehicle emissions. Ground transportation accidents would be expected to result in no additional radiological impacts to the population in the vicinity of the accident. Potential impacts from low probability accidents vary considerably and are dependent on the accident conditions (such as the size of the resulting fire, if any) and the weather conditions at the time of an accident. Transportation accidents were estimated to result in no LCF due to radiation and less than 0.2 immediate deaths due to traffic fatalities (see Section 4.2.3.3).

As described in Appendix A, the percentage of the total population comprised of minorities or low-income households varies among routes. Calculations for incident-free and accident conditions demonstrate that for the general population the radiological impacts would be very low. Minority or low-income populations living near these routes would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. They would be subject to very low impacts as would the general population.

Implementation of the proposed action would have extremely low nonradiological effects on the environment along transportation routes, including the social and economic status of the general population, minority populations, and the low-income population residing along the transportation routes. Economic benefits that would result from increased transportation of cargo along transportation routes would be extremely small for the general population or any particular segment of the population residing along the transportation routes.

4.2.4 Foreign Research Reactor Spent Nuclear Fuel Management Sites

This section presents the potential environmental impacts from the basic implementation of Management Alternative 1 at the potential foreign research reactor spent nuclear fuel management sites, namely the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site. It summarizes the detailed site analysis presented in Appendix F, Sections F.4, F.5, and F.6. The analysis examined environmental topics such as land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, occupational and public health and safety, noise, traffic and transportation, utilities and energy, and waste management. The analysis showed that the basic implementation of Management Alternative 1 would not have a major effect on any of the environmental topics. Further, none of the environmental topics would clearly differentiate among the potential foreign research reactor spent nuclear fuel management sites.

Because of the public interest in radiation exposure to workers and the public, Section 4.2.4.1 discusses in detail the impacts on occupational and public health and safety from the basic implementation of Management Alternative 1, even though the analysis concludes that such impacts are very low. Section 4.2.4.2 summarizes the impacts on the other environmental topics. Section 4.2.4.3 discusses the cumulative impacts of the basic implementation of Management Alternative 1 at each candidate management site, and Section 4.2.4.4 addresses the waste management and mitigation measures available under the basic implementation of Management Alternative 1. Later in this chapter, Section 4.10 compares the risks of the basic implementation of Management Alternative 1 to risks of common activities.

4.2.4.1 Occupational and Public Health and Safety

Possible sources of occupational and public radiological exposure from foreign research reactor spent nuclear fuel include: (1) emissions of radioactive material from incident-free operations, (2) incident-free handling activities, and (3) emissions from accident conditions. Foreign research reactor spent nuclear fuel management is not expected to impact occupational and public health and safety. Nonradiological exposures are not likely to occur during construction or operation of foreign research reactor spent nuclear fuel storage facilities. Radiological exposures are presented in individual subsections for emissions-related impacts, handling-related impacts, and accident-related impacts.

Conservative Assumptions and Impacts to the Public of Incident-Free Site Activities

Doses that could be received by the public during incident-free operation of foreign research reactor spent nuclear fuel storage facilities could only be due to emissions of radioactive material that becomes airborne. The public would be too far from the storage facilities to receive any direct exposure. In summary:

- Doses were calculated for the MEI, defined as an individual living at the management site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the facility. These doses would result from incident-free airborne radiological emissions released during foreign research reactor spent nuclear fuel transfer from the transportation cask to the storage facility and from foreign research reactor spent nuclear fuel storage.

- Radiological airborne emissions consist of two parts: (1) emissions from gaseous releases during receipt and unloading of the transportation casks; and (2) emissions during the management period. The emissions during receipt and unloading were calculated conservatively assuming one percent of the foreign research reactor spent nuclear fuel would fail during transport and the associated gaseous fission products would be released during the transfer at the management site. DOE and the Department of State also conservatively assumed that unloading the spent nuclear fuel cask in a dry cell would allow all free gaseous fission products to be released to the environment, while unloading in a wet pool would allow 90 percent of the halogens to be retained in the water. Radiological emissions during wet storage were based on historical data at the Receiving Basin for Offsite Fuels (RBOF) at the Savannah River Site. The emissions during incident-free dry storage would be zero because the spent nuclear fuel would be stored in sealed containers. The methodology and conservative assumptions used for the calculation of radiological emissions under the basic implementation of Management Alternative 1 are discussed in detail in Appendix F, Section F.6.

- Doses were calculated separately for each phase of the program at each candidate management site to accommodate the two-phased implementation of the basic implementation of Management Alternative 1. For example, in the case where the Nevada Test Site, the Hanford Site, or the Oak Ridge Reservation is selected as a Phase 2 site, with the Savannah River Site or the Idaho National Engineering Laboratory as a Phase 1 site, doses were calculated at the Savannah River Site or the Idaho National Engineering Laboratory for Phase 1, and at the Hanford Site, Oak Ridge Reservation, or the Nevada Test Site for Phase 2.

- Doses from an operation which combines an existing wet or dry storage facility for spent nuclear fuel receiving and characterization and dry storage casks to enhance storage capacity are bounded by the doses calculated for the existing facility.

- Doses were conservatively calculated for the maximum quantity of foreign research reactor spent nuclear fuel that could be received at each storage site as discussed in Appendix F, Section F.4.

Tables 4-8 through 4-12 summarize the annual emission-related doses to the public and the associated risks for the MEI and population at each site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period. In general, receipt and unloading at wet storage facilities produces lower public risk than at dry storage facilities.

Table 4-8 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Savannah River Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• RBOF (wet storage)	0.00011	5.5×10^{-11}	0.0057	0.0000028
• L-Reactor Basin (wet storage)	0.000073	3.7×10^{-11}	0.0046	0.0000023
• New Dry Storage Facility	0.00018	9.0×10^{-11}	0.0086	0.0000043
Storage at:				
• RBOF (wet storage)	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
• L-Reactor Basin (wet storage)[a]	0.00036	1.8×10^{-10}	0.022	0.000011
• New Dry Storage Facility	0	0	0	0

[a] *L-Reactor basin doses are due to existing conditions. The foreign research reactor spent nuclear fuel contribution would be six orders of magnitude lower.*

Table 4-9 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Idaho National Engineering Laboratory

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• IFSF[a]/CPP-749 (dry storage)	0.00056	2.8×10^{-10}	0.0045	0.0000023
• Fluorinel Dissolution and Fuel Storage (FAST) (wet storage)	0.00038	1.9×10^{-10}	0.0031	0.0000016
• New Dry Storage Facility[b]	0.00056	2.8×10^{-10}	0.0045	0.0000023
Storage at:				
• IFSF[a]/CPP-749 (dry storage)	0	0	0	0
• FAST (wet storage)	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
• New Dry Storage Facility[b]	0	0	0	0

[a] *Irradiated Fuel Storage Facility*

[b] *The doses for this new dry storage facility are assumed to be equal to those for IFSF/CPP-749.*

Table 4-10 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Hanford Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• Fuel Material Examination Facility (FMEF) (dry storage)	0.00020	1.0×10^{-10}	0.011	0.0000055
• New Dry Storage Facility[a]	0.00025	1.3×10^{-10}	0.015	0.0000075
Storage at:				
• FMEF (dry storage)	0	0	0	0
• New Dry Storage Facility[a]	0	0	0	0

[a] *The doses for this new dry storage facility are different from those for FMEF due to the different release height and location.*

Table 4-11 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Oak Ridge Reservation

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• New Dry Storage Facility	0.089	4.5×10^{-8}	0.085	0.000043
Storage at:				
• New Dry Storage Facility	0	0	0	0

Table 4-12 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• Engine Maintenance and Disassembly (E-MAD) (dry storage)	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
• New Dry Storage Facility[a]	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
Storage at:				
• E-MAD (dry storage)	0	0	0	0
• New Dry Storage Facility[a]	0	0	0	0

[a] *The doses for this new dry storage facility are assumed to be equal to those for E-MAD.*

Among all the potential foreign research reactor spent nuclear fuel management sites, the maximum estimated annual incident-free public MEI radiological exposure from emissions is 0.09 mrem per year. This exposure would occur at the Oak Ridge Reservation (Table 4-11) during receipt and handling. It is much higher than all other corresponding dose rates in Tables 4-8 through 4-12. The receipt period would be about 3 years, so the total MEI dose would be 0.27 mrem. The associated probability for incurring one LCF would be 1.4×10^{-7} for the MEI, which represents less than two chances in ten million of developing a fatal cancer from radiological exposure.

The highest annual incident-free population risk among the Savannah River Site and the Idaho National Engineering Laboratory (Phase 1 sites) is 0.000011 LCF per year (Tables 4-8 and 4-9), which would be due to emissions from L-Reactor Basin at the Savannah River Site. Assuming some foreign research reactor spent nuclear fuel is stored in this basin for the entire 10 years of Phase 1 plus 3 years to transfer it to a Phase 2 site, the Phase 1 component of this population risk would be as high as 0.00014 LCF. The highest annual incident-free population risk from a new dry storage facility at a potential Phase 2 site (Tables 4-8 through 4-12), is 0.000043 LCF per year, which would be due to receipt/unloading at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be received at the Oak Ridge Reservation for as long as 3 years, the Phase 2 component of this population risk would be 0.00013 LCF. This is higher than any other combination of Phase 2 dry storage annual risks and durations. Adding the Phase 1 and Phase 2 population risks yields 0.00027 LCF for the total population risk to the public living near the sites due to incident-free conditions.

Conservative Assumptions and Impacts to Workers of Incident-Free Site Activities

Workers would receive radiation doses during handling operations, such as receiving and unloading foreign research reactor spent nuclear fuel transportation casks at the management site, transferring foreign research reactor spent nuclear fuel from one facility to another within the management site, or packaging the foreign research reactor spent nuclear fuel for shipment to another management site. Detailed analysis of the potential impacts is given in Appendix F of this EIS. In summary:

- The maximally exposed worker dose estimate is based on the regulatory limit of 5,000 mrem per year for radiation workers at all DOE management sites. DOE and the Department of State conservatively assumed that an individual worker received this dose every year for all 13 years that the handling operations would be in progress. Although this assumption is highly unlikely, the calculated total maximally exposed worker dose is 65,000 mrem and the associated risk is 0.026 LCF. This means that this individual would have a nearly three percent higher chance of incurring an LCF.

- Worker population doses were estimated by considering the type and duration of all operations performed by the workers during the handling of each transportation cask and storage cask as appropriate, including: (1) the number of workers needed, (2) the duration of a specific operation, and (3) the distance between the transportation cask and the operation being performed. Only the workers actually performing the operations receive radiation doses, and thus would have an increased risk of incurring an LCF. If the total radiation dose is received by a small number of workers, each worker would have a higher risk of cancer than if the total dose is received by a large number of workers. The dose rate in the vicinity of the transportation and storage casks assumed for the estimates was based on the conservative methodology presented in Appendix F, Section F.5. As noted in Section F.5, worker population doses associated with dry storage cask design may be higher than those associated with the vault design because of the additional worker activities associated with the handling of the cask that transfers the canistered spent fuel to the concrete structure. The worker population doses reported below for new dry storage conservatively reflect the cask design.

- The number of casks handled at each potential foreign research reactor spent nuclear fuel management site would depend on the number of cask shipments considered under the ground transportation options discussed in Section 2.6.4.1, and the amount of foreign research reactor spent nuclear fuel expected to be transferred between facilities during Phase 2.

Table 4-13 provides a summary of the number of casks that would be handled at each potential foreign research reactor spent nuclear fuel management site under the Centralization Alternative in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and in the current EIS.

Table 4-13 Estimated Number of Shipments to and from Each Potential Foreign Research Reactor Spent Nuclear Fuel Management Site

Candidate Storage Site	*Incoming Shipments*	*Intersite Shipments*	*Outgoing Shipments*	*Total Shipments*
Savannah River Site or Idaho National Engineering Laboratory Phase 1	644[a]	0	161	805
Savannah River Site or Idaho National Engineering Laboratory Phases 1 and 2	837[b]	209	0	1,046
Hanford Site or Oak Ridge Reservation or Nevada Test Site Phase 2	354[c]	0	0	354

[a] *10-year receipt in foreign research reactor spent nuclear fuel certified casks.*

[b] *13-year receipt in foreign research reactor spent nuclear fuel certified casks.*

[c] *161 from near term site using large truck casks and 193 from ports using foreign research reactor spent nuclear fuel certified casks.*

Tables 4-14 through 4-18 present the population doses and risks that would be received by the members of the working crew, if that crew handled the total number of casks at each management site. The results do not include shipments in large rail casks.

Table 4-14 Handling-Related Impacts to Workers at the Savannah River Site

	Worker Population Dose (person-rem)		*Worker Population Risk (LCF)*	
	RBOF/L-Reactor	*New Dry Storage*	*RBOF/L-Reactor*	*New Dry Storage*
Phase 1	250	NA	0.10	NA
Phases 1 and 2	NA	416[a]	NA	0.17[a]

[a] *Cask design*

Table 4-15 Handling-Related Impacts to Workers at the Idaho National Engineering Laboratory

	Worker Population Dose (person-rem)			*Worker Population Risk (LCF)*		
	IFSF[a]/CPP-749	*FAST*	*New Dry Storage*	*IFSF[a]/CPP-749*	*FAST*	*New Dry Storage*
Phase 1	257	250	NA	0.10	0.10	NA
Phases 1 and 2[b]	NA	NA	424[c]	NA	NA	0.17[c]
Phases 1 and 2[d]	NA	NA	416[c]	NA	NA	0.17[c]

[a] *Irradiated Fuel Storage Facility*

[b] *Phase 1 at IFSF/CPP-749*

[c] *Cask design*

[d] *Phase 1 at FAST*

Table 4-16 Handling-Related Impacts to Workers at the Hanford Site

	Worker Population Dose (Person-rem) FMEF/New Dry Storage	Worker Population Risk (LCF) FMEF/New Dry Storage
Phase 2	266[a]	0.11[a]

[a] *Cask design*

Table 4-17 Handling-Related Impacts to Workers at the Oak Ridge Reservation

	Worker Population Dose (Person-rem) New Dry Storage	Worker Population Risk (LCF) New Dry Storage
Phase 2	266[a]	0.11[a]

[a] *Cask design*

Table 4-18 Handling-Related Impacts to Workers at the Nevada Test Site

	Worker Population Dose (person-rem)		Worker Population Risk (LCF)	
	E-MAD	New Dry Storage	E-MAD	New Dry Storage
Phase 2	113	266[a]	0.05	0.11[a]

[a] *Cask design*

According to the above tables, the highest dose to a working crew at a single site would be 424 person-rem at the Idaho National Engineering Laboratory in the analyzed case which assumes that all foreign research reactor spent nuclear fuel is received in the Irradiated Fuel Storage Facility and/or the CPP-749 facility (dry storage) during Phase 1 and is transferred to a new dry storage facility at the Idaho National Engineering Laboratory in Phase 2. The associated number of additional LCF is 0.17. The highest dose to working crews for both phases in more than one site is 523 person-rem: 266 person-rem at one of the 3 Phase 2 sites, plus 257 person-rem at the Idaho National Engineering Laboratory as the Phase 1 site. The associated probability for developing one LCF among the working crews of the two sites is 0.21.

Conservative Assumptions and Accident-Related Impacts

An evaluation of hypothetical accidental radioactive material releases at the potential foreign research reactor spent nuclear fuel management sites was performed to assess the impact of possible radiation exposure to individuals and the general population (see also Appendix F, Section F.6). All inputs are site-specific except for the radioactivity release. Site-specific information includes meteorological conditions, population distribution, and food production and consumption rates within 80 km (50 mi) of the management location.

The radiation doses to the following individuals and the general population are calculated for accident conditions at the spent nuclear fuel management facility:

- Worker: An individual located 100 m (330 ft) from the radioactive material release point. (The impact of accidents on close-in workers is not calculated numerically but is discussed qualitatively for each accident at the end of this section.) For elevated release (from a tall stack), the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose occurs is frequently greater than 100 m (330 ft) for elevated release.

The direction to the worker was chosen as the direction to the maximally exposed sector. The dose to the worker is calculated for the 50th-percentile meteorological condition (DOE, 1992a).

- Maximally Exposed Individual (MEI): A theoretical member of the general public living at the management site boundary receiving the maximum exposure. This individual is conservatively assumed to be located in a direction downwind from the release point. The dose to the MEI is shown for the conservative 95th-percentile meteorological condition.

- Nearest Public Access Individual (NPAI): An individual stranded on a highway or public access road near to the facility at the time of an accident. The distance to the NPAI was chosen as the distance to the nearest public access point; the direction was chosen as the direction to that point. The dose to the NPAI is shown for the conservative 95th-percentile meteorological condition.

- General population within an 80-km (50-mi) radius of the facility: The dose calculations are performed for the direction downwind from the release point that results in highest dose to the public. The dose to the population is shown for the conservative 95th-percentile meteorological condition.

The radiation dose to individuals and the public resulting from exposure to radioactive contamination was calculated using external (direct exposure), inhalation, and ingestion pathways. Dispersion in air from point of release was estimated with both 50th-percentile and 95th-percentile meteorological conditions. The 50th-percentile condition represents the median meteorological condition. The 95th-percentile condition is defined as that condition which is not exceeded more than 5 percent of the time, and is more conservative than the 50th-percentile condition.

The ingestion dose is calculated by considering that the individual and the public would consume the contaminated food produced in the vicinity [up to 80 km (50 mi)] of the accident. This is conservative, and it is expected that continued consumption of contaminated food products by the public would be suspended if the projected dose exceeded the protective action guidelines developed by the U.S. Environmental Protection Agency (EPA, 1991a). To ensure a consistent and conservative analytical basis, no reduction of exposure due to a protective action guideline was used in this analysis.

Accidents considered for detailed analysis are similar to those that were analyzed in the Programmatic SNF&INEL Final EIS. The selection of the accidents was based on the following considerations:

(1) criticality caused by human error during operation, equipment failure, or earthquake; (2) mechanical damage to foreign research reactor spent nuclear fuel during examination and preparation (cropping off the aluminum and nonfuel end of a spent fuel element); and (3) accident involving an impact by either an internal or an external initiator with and without an ensuing fire.

Six accident scenarios were evaluated at each management location using identical source terms (estimated amounts of radioactive material released during postulated accidents). The wet pool accidents are assumed to be cutting into the fuel region or mechanical damage due to operator error, an accidental criticality, and an aircraft crash into the water pool facility. The dry storage accidents are assumed to be cutting into the fuel region or mechanical damage during examination work and handling in a dry cell, dropping of a spent nuclear fuel cask, and an aircraft crash with an ensuing fire.

Tables 4-19 through 4-23 present the frequencies and the consequences of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the conservative assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE and the Department of State did not estimate the worker population dose due to accidents.

Table 4-19 Frequency and Consequences of Accidents at the Savannah River Site

	Frequency (per yr)	*Consequences*							
		MEI		*NPAI*		*Population*		*Worker*	
		(mrem)	*(LCF)*	*(mrem)*	*(LCF)*	*(person-rem)*	*(LCF)*	*(mrem)*	*(LCF)*
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	0.24	1.2×10^{-7}	0.068	3.4×10^{-8}	9.2	0.0046	28	0.000011
Dropped Spent Nuclear Fuel Cask	0.0001	0.018	9.0×10^{-9}	0.00034	1.7×10^{-10}	0.55	0.00028	0.28	1.1×10^{-7}
Aircraft Crash w/Fire	1×10^{-6}	40	0.00002	0.29	1.5×10^{-7}	1300	0.65	120	0.000048
Wet Storage Accidents - RBOF									
Spent Nuclear Fuel Assembly Breach	0.16	0.0070	3.5×10^{-9}	0.00039	2.0×10^{-10}	0.23	0.00012	0.14	5.6×10^{-8}
Accidental Criticality	0.0031	130	0.000065	44	0.000022	4,800	2.4	16,000	0.0064
Aircraft Crash	1×10^{-6}	4.1	0.0000021	0.98	4.9×10^{-7}	150	0.075	400	0.00016
Wet Storage Accidents - L-Reactor Basin									
Spent Nuclear Fuel Assembly Breach	0.16	0.0093	4.7×10^{-9}	0.00097	4.9×10^{-10}	0.14	0.00007	0.11	4.4×10^{-8}
Accidental Criticality	0.0031	170	0.000085	120	0.000060	3,000	1.5	14,000	0.0056
Aircraft Crash	1×10^{-6}	4.2	0.0000021	2.6	0.0000013	93	0.047	70	0.000028

[a] *New Dry Storage Facility*

Table 4-20 Frequency and Consequences of Accidents at the Idaho National Engineering Laboratory

	Frequency (per yr)	*Consequences*							
		MEI		*NPAI*		*Population*		*Worker*	
		(mrem)	*(LCF)*	*(mrem)*	*(LCF)*	*(person-rem)*	*(LCF)*	*(mrem)*	*(LCF)*
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	1.3	6.5×10^{-7}	0.67	3.4×10^{-7}	15	0.0075	28	0.000011
Dropped Spent Nuclear Fuel Cask	0.0001	0.074	3.7×10^{-8}	0.0033	1.7×10^{-9}	0.83	0.00042	0.12	4.8×10^{-8}
Aircraft Crash w/Fire	1×10^{-6}	180	0.00009	2.9	0.0000015	2,000	1.0	120	0.000048
Wet Storage Accidents									
Spent Nuclear Fuel Assembly Breach	0.16	0.0016	8.0×10^{-10}	0.0036	1.8×10^{-9}	0.43	0.00022	0.14	5.6×10^{-8}
Accidental Criticality	0.0031	28	0.000014	30	0.000015	140	0.070	1800	0.00072
Aircraft Crash	1×10^{-6}	22	0.000011	9.8	0.0000049	250	0.13	400	0.00016

[a] *New Dry Storage Facility at IFSF/CPP-749*

Table 4-21 Frequency and Consequences of Accidents at the Hanford Site

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	3.0	0.0000015	0.57	2.9×10^{-7}	42	0.021	50	0.000020
Dropped Spent Nuclear Fuel Cask	0.0001	0.26	1.3×10^{-7}	0.0085	4.3×10^{-9}	3.0	0.0015	0.22	8.8×10^{-8}
Aircraft Crash w/Fire[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
Dry Storage Accidents at FMEF									
Spent Nuclear Fuel Assembly Breach[c]	0.16	4.7	0.0000024	2.1	0.0000011	46	0.023	0.99	4.0×10^{-7}
Dropped Spent Nuclear Fuel Cask[c]	0.0001	0.2	1.0×10^{-7}	0.032	1.6×10^{-8}	3.2	0.0016	0.0049	2.0×10^{-9}
Aircraft Crash w/Fire[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA

[a] *New Dry Storage Facility*

[b] *Aircraft Crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

[c] *Emissions would be released through a tall stack, so workers would receive low doses.*

NA = Not applicable

Table 4-22 Frequency and Consequences of Accidents at the Oak Ridge Reservation

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	22	0.000011	42	0.000021	55	0.028	140	0.000056
Dropped Spent Nuclear Fuel Cask	0.0001	1.4	7.0×10^{-7}	0.18	9.0×10^{-8}	15	0.0075	0.61	2.4×10^{-7}
Aircraft Crash w/Fire	1×10^{-6}	2300	0.0012	180	0.000090	2900	1.5	610	0.00024

[a] *New Dry Storage Facility*

Table 4-23 Frequency and Consequences of Accidents at the Nevada Test Site

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	1.7	8.5×10^{-7}	0.31	1.6×10^{-7}	1.5	0.00075	20	0.0000080
Dropped Spent Nuclear Fuel Cask	0.0001	0.11	5.5×10^{-8}	0.0014	7.0×10^{-10}	0.40	0.00020	0.089	3.6×10^{-8}
Aircraft Crash w/Fire	1×10^{-6}	180	0.000090	1.2	6.0×10^{-7}	250	0.13	87	0.000035

[a] *E-MAD and New Dry Storage Facility*

The analyses were performed for a generic dry storage at the five potential foreign research reactor spent nuclear fuel management sites, as well as for site-specific locations (i.e., FMEF at the Hanford Site, E-MAD at the Nevada Test Site, L-Reactor Basin and RBOF at the Savannah River Site).

Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at each candidate management site. These annual risk estimates are presented in Tables 4-24 through 4-28.

Table 4-24 Annual Risks of Accidents at the Savannah River Site

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	1.9×10^{-8}	5.5×10^{-9}	0.00075	0.0000018
Dropped Spent Nuclear Fuel Cask	9.0×10^{-13}	1.7×10^{-14}	2.8×10^{-8}	1.1×10^{-11}
Aircraft Crash w/Fire	2.0×10^{-11}	1.5×10^{-13}	6.5×10^{-7}	4.8×10^{-11}
Wet Storage Accidents at RBOF				
Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
Accidental Criticality	2.0×10^{-7}	7.0×10^{-8}	0.0074	0.000020
Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
Wet Storage Accidents at L-Reactor Basin				
Spent Nuclear Fuel Assembly Breach	7.4×10^{-10}	8.0×10^{-11}	0.000011	7.1×10^{-9}
Accidental Criticality	2.6×10^{-7}	1.9×10^{-7}	0.0047	0.000017
Aircraft Crash	2.1×10^{-12}	1.3×10^{-12}	4.7×10^{-8}	2.8×10^{-11}

[a] *New Dry Storage Facility.*

Table 4-25 Annual Risks of Accidents at the Idaho National Engineering Laboratory

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	1.1×10^{-7}	5.5×10^{-8}	0.0012	0.0000018
Dropped Spent Nuclear Fuel Cask	3.7×10^{-12}	1.7×10^{-13}	4.2×10^{-8}	4.8×10^{-12}
Aircraft Crash w\Fire	9.0×10^{-11}	1.5×10^{-12}	0.0000010	4.8×10^{-11}
Wet Storage Accidents[b]				
Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}

[a] *IFSF/CPP-749 or New Dry Storage Facility*

[b] *FAST Facility*

The highest annual MEI or NPAI risk among the potential Phase 1 sites (Tables 4-24 and 4-25) is 2.6×10^{-7} LCF per year, which is the annual risk to the MEI from accidental criticality at L-Reactor Basin. Assuming some foreign research reactor spent nuclear fuel is stored in RBOF for the entire 10 years of Phase 1 plus 3 years to transfer it to a Phase 2 site, the Phase 1 component of this MEI risk would be 0.0000034 LCF. The highest annual MEI or NPAI risk due to dry storage facility accidents at the potential Phase 2 sites (Tables 4-24 through 4-28) is 0.0000034 LCF per year, which is the annual risk to the NPAI from an assembly breach accident during handling at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be received at the Oak Ridge Reservation for as long as 3 years, the Phase 2 component of this MEI/NPAI risk would be 0.000010 LCF. This is higher than any other

Table 4-26 Annual Risks of Accidents at the Hanford Site

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	2.4×10^{-7}	4.6×10^{-8}	0.0034	0.0000032
Dropped Spent Nuclear Fuel Cask	1.3×10^{-11}	4.3×10^{-13}	1.5×10^{-7}	8.8×10^{-12}
Aircraft Crash w\Fire[b]	NA	NA	NA	NA
Dry Storage Accidents at FMEF				
Spent Nuclear Fuel Assembly Breach[c]	3.7×10^{-7}	1.7×10^{-7}	0.0037	6.4×10^{-8}
Dropped Spent Nuclear Fuel Cask[c]	8.0×10^{-12}	1.6×10^{-12}	1.6×10^{-7}	2.5×10^{-13}
Aircraft Crash with Fire[b]	NA	NA	NA	NA

[a] *New Dry Storage Facility*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

[c] *Emissions would be released through a tall stack.*

NA = Not applicable

Table 4-27 Annual Risks of Accidents at the Oak Ridge Reservation

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	0.0000018	0.0000034	0.0044	0.0000088
Dropped Spent Nuclear Fuel Cask	7.0×10^{-11}	9.0×10^{-12}	7.5×10^{-7}	2.4×10^{-11}
Aircraft Crash w\Fire	1.2×10^{-9}	9.0×10^{-11}	0.0000015	2.4×10^{-10}

[a] *New Dry Storage Facility*

Table 4-28 Annual Risks of Accidents at the Nevada Test Site

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	1.4×10^{-7}	2.5×10^{-8}	0.00012	0.0000013
Dropped Spent Nuclear Fuel Cask	5.5×10^{-12}	7.0×10^{-14}	2.0×10^{-8}	3.6×10^{-12}
Aircraft Crash w\Fire	9.0×10^{-11}	6.0×10^{-13}	1.3×10^{-7}	3.5×10^{-11}

[a] *E-MAD and New Dry Storage Facility*

combination of Phase 2 annual accident risks and associated durations. Taking the maximum of the Phase 1 and Phase 2 MEI risks yields 0.000010 LCF for the maximum MEI risk due to accidents. This means that the MEI has one chance in one hundred thousand of incurring an LCF due to accidents.

The highest annual population risk among the potential Phase 1 sites (Tables 4-24 and 4-25) is 0.0074 LCF per year, which is the annual population risk from an accidental criticality at RBOF. Assuming some foreign research reactor spent nuclear fuel is stored in RBOF for the entire 10 years of Phase 1, plus 3 years to transfer it to a Phase 2 site, the Phase 1 component of this population risk would be 0.096 LCF. The highest annual population risk due to dry storage facility accidents at the potential Phase 2 sites (Tables 4-24 through 4-28) is 0.0044 LCF per year, which is the annual risk to the public from assembly breach accidents during handling at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be received at the Oak Ridge Reservation for as long as 3 years, the Phase 2 component of this population risk would be 0.013 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations. Adding the Phase 1 and Phase 2 population risks yields 0.11 LCF for the total population risk due to accidents.

Impacts of Accidents on Close-in Workers

An evaluation has been made of the impacts to close-in workers involved in spent fuel handling and management operations. This evaluation focuses on the radiological consequences of the accident. Clearly, a limited number of fatalities could occur which would be related to spent nuclear fuel handling only in an indirect or secondary manner (e.g., the worker who happened to be in the facility might be killed due to an aircraft crash).

Wet Storage Accidents

Fuel Assembly Breach in Wet Storage: No fatalities of nearby workers would be expected due to radiological consequences. This is because the release of the radionuclides would be underwater. Attenuation by the water would occur for most of the release products; release of the noble gases from the pool would, however, cause a direct radiation exposure to workers in the area. If radiation is released from the surface of the water pool, radiation alarms would sound, prompting evacuation of nearby workers.

Dropped Fuel Cask in Wet Storage: No fatalities would be expected due to the radiological consequences of this accident. The operation of the crane is done using remote controls, so workers are not likely to be in the direct vicinity of the dropped cask.

Accidental Criticality: The accidental criticality could occur at a minimum of 3 m (10 ft) underwater. Based on the shielding provided by the water pool, it is likely that no fatalities would occur. Nearby workers would likely receive appreciable radiation exposures.

Aircraft Crash into the Water Pool: No fatalities to nearby workers would be expected due to radiological consequences. An aircraft crash into the water pool would prompt nearby workers not affected by the crash to evacuate the area immediately. The release of radiation products would be underwater, allowing sufficient time for evacuation before radiation products would reach the surface.

Dry Storage Accidents

Fuel Assembly Breach: Cropping of the fuel assembly would occur in a dry cell. Any release that would occur due to inadvertent cutting into the fuel would be confined to the storage cell, where the exhaust is away from the workers. No fatalities to nearby workers would be expected due to this scenario.

Dropped Fuel Cask: No fatalities would be expected due to the radiological consequences of this accident. The operation of the crane is done using remote controls, so workers are not likely to be in the direct vicinity of the dropped cask. In addition, the workers would promptly leave the area.

Aircraft Crash with Fire: If an aircraft crashes into a dry storage facility and catches fire, large amounts of radioactive material could be released into the atmosphere. If the facility is occupied at the time of the crash, any surviving workers could receive a substantial radiation dose from the released radioactive material.

Secondary Impacts of Accidents

Impacts of accidents on resources other than human health and safety (secondary impacts), have been addressed in Section F.4 for each management site. The general conclusion is that no measurable secondary impacts to land uses, cultural resources, water quality, ecological resources, national defense, or local economies are expected from the postulated accidents involving foreign research reactor spent nuclear fuel at the management sites.

4.2.4.2 Topics Not Discussed in Detail

This section summarizes the potential impacts for the environmental topics not covered in Section 4.2.4.1, namely land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, noise, utilities and energy, and waste management. The detailed analysis of these topics presented in Appendix F, Section F.4 showed that none of these topics clearly differentiated among the potential foreign research reactor spent nuclear fuel management sites nor had any major environmental impact. The discussion of each topic generally concentrates on management sites and alternatives that have the largest estimated impacts, and demonstrates that the environmental impacts for that topic are not of sufficient magnitude to be given strong consideration in the decision making process.

4.2.4.2.1 Land Use

The basic implementation of Management Alternative 1 would only result in minor land use impacts at any of the potential foreign research reactor spent nuclear fuel management sites. The largest land use impact would be 16 ha (40 acres) at the Oak Ridge Reservation to construct a new dry storage facility. This represents less than 0.1 percent of the total size of the Oak Ridge Reservation. A description of the land use impacts at the other potential foreign research reactor spent nuclear fuel management sites is contained in Appendix F.4. For all of the potential foreign research reactor spent nuclear fuel management sites, new foreign research reactor spent nuclear fuel storage facilities would be built on land previously disturbed or designated for industrial use. No additional land outside of the existing sites would be required for foreign research reactor spent nuclear fuel management. It should be noted that land use and other environmental impacts associated with the construction activities would be minimal, under the implementation alternatives that use refurbishment of existing facilities for interim storage (i.e., BNFP at the Savannah River Site and E-MAD at the Nevada Testing Site). All environmental impacts from the refurbishment and operation of these facilities would be bounded by the impacts associated with the construction and operation of new generic storage facilities. Land use impacts are discussed in more detail in Appendix F, Section F.4.

4.2.4.2.2 Socioeconomics

The basic implementation of Management Alternative 1 would only result in minor socioeconomic impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Socioeconomic impacts are defined for purposes of this analysis in terms of direct effects, which include changes in site employment and expenditures from foreign research reactor spent nuclear fuel-related construction and operation and indirect effects, such as changes that result from regional purchases, nonpayroll expenditures, and payroll spending by site employees.

No construction personnel would be needed for existing facilities, and not more than 240 workers per year (peak) would be needed to build a new dry storage facility. The annual staffing requirements for operations would be about 30 and 8 full-time employees during receipt and storage, respectively, for a new dry storage facility. This would represent 0.15 to 0.9 percent of the existing work force at any of the potential foreign research reactor spent nuclear fuel management sites. No new hiring would be expected because most positions would be filled by reassignments of the existing work force. Even if all operational positions were filled by new hires, this would represent about an even smaller increase in regional employment. The secondary effects would be even lower.

4.2.4.2.3 Cultural Resources

The basic implementation of Management Alternative 1 would only result in minor cultural impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Cultural, archaeological, historic, and architectural resources are defined as prehistoric and historic sites, districts, structures, and evidence of human use that are considered to be important to a culture, subculture, or a community for scientific, traditional, religious, or other reasons.

Although most of the potential foreign research reactor spent nuclear fuel management sites contain areas of archaeological, cultural, or historical interest, little or no direct impacts on cultural resources would be expected because of the location of the foreign research reactor spent nuclear fuel storage facilities. Specific site surveys have not been completed; however, based on existing information, no known cultural resources would be affected by construction or operation of foreign research reactor spent nuclear fuel facilities. Prior to construction, specific site surveys would be conducted. In the event that cultural resources were encountered during construction, the State Historic Preservation Officer would be contacted immediately. Similarly, Tribal leaders would be notified if any Native American resources were found.

4.2.4.2.4 Aesthetic and Scenic Resources

The basic implementation of Management Alternative 1 would only result in minor impacts to aesthetic and scenic resources at any of the potential foreign research reactor spent nuclear fuel management sites. Foreign research reactor spent nuclear fuel storage facilities would be located far from public view in areas previously disturbed or designated for industrial use. Construction activities would generate fugitive dust that could temporarily affect visibility. However, best management practices would be implemented to minimize such conditions. Furthermore, facility operations would not produce emissions that would adversely impact visibility.

4.2.4.2.5 Geology

The basic implementation of Management Alternative 1 would only result in minor geologic impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Except for the potential existence of gold, tungsten, and molybdenum at the Nevada Test Site, geologic resources consist only of surficial sand, gravel, or clay deposits, all of which have low economic value. Construction activities would disturb these surface deposits, but because of the large volume of these materials on the potential foreign research reactor spent nuclear fuel management sites, the impact would be expected to be small.

4.2.4.2.6 Air Quality

The basic implementation of Management Alternative 1 would only result in minor impacts on air quality at any of the potential foreign research reactor spent nuclear fuel management sites. The projected emissions from foreign research reactor spent nuclear fuel storage at the potential management sites would not contribute to Federal or State nonattainment standards. Construction activities would be expected to cause only temporary, minor increases in fugitive dust emissions, but the use of standard dust suppression techniques would be expected to mitigate this problem. Particulate emissions could temporarily affect visibility in localized areas, but would not adversely affect Federal or State attainment standards.

4.2.4.2.7 Water Quality

The basic implementation of Management Alternative 1 would have only minor impacts on water resources at the potential foreign research reactor spent nuclear fuel management sites. Water consumption during construction would require very small amounts of water when compared to daily water usage at the potential management sites.

During operations, the greatest amount of water consumed annually would be about 2.1 million liters (550,000 gal) per year. This amount represents no more than 0.2 percent of the annual water consumption at any of the potential foreign research reactor spent nuclear fuel management sites. At the Nevada Test Site, where available water is limited, a cumulative water supply impact could be important from activities other than foreign research reactor spent nuclear fuel management, but the foreign research reactor spent nuclear fuel management contribution would be very small. Further study of the Ash Meadows sub-basin would be required to specify the exact impact on aquifer yield and integrity.

Under normal operations there would be no direct discharge of effluent to ground or surface waters from a new dry storage facility.

4.2.4.2.8 Ecology

The basic implementation of Management Alternative 1 would only result in minor ecological impacts at the potential foreign research reactor spent nuclear fuel management sites. Under any construction of new facilities, individual or small populations of some wildlife species could be disturbed, displaced, or destroyed. However, the size of the areas affected would be small in relation to the size of the potential foreign research reactor spent nuclear fuel management sites and the size of remaining natural habitats. The type of habitats affected could vary but would be typical of the regional area in which the foreign research reactor spent nuclear fuel storage facility is located. For this reason, any such habitat losses would probably not affect any threatened or endangered species or critical habitats in the area. Habitat fragmentation is not expected because new storage facilities would be constructed on land that has been previously disturbed or designated for industrial purposes. Mitigation plans would be developed in consultation with the appropriate agencies if any threatened or endangered species were identified.

DOE has begun or has completed consultations with the U.S. Fish and Wildlife Service regarding threatened and endangered species for the proposed construction site of foreign research reactor spent nuclear fuel storage facilities at the five potential sites, as required by the Endangered Species Act.

4.2.4.2.9 Noise

The basic implementation of Management Alternative 1 would only result in minor noise impacts at the potential foreign research reactor spent nuclear fuel management sites. Construction activities would generate noise levels consistent with light industrial activity. Based on existing studies these noises would not be expected to propagate offsite at levels that would affect the general population. Noises generated during operations would be less than those during construction.

4.2.4.2.10 Materials, Utilities, and Energy

The basic implementation of Management Alternative 1 would only result in minor impacts on materials, utilities, and energy at the potential foreign research reactor spent nuclear fuel management sites. For existing facilities, incremental increases in materials, utilities, and energy would be very small. New dry storage facilities would result in increased demands on water, power, and sewage. The increased water usage during construction would add no more than 0.2 percent to existing sitewide levels. Increased annual electricity requirements would be about 800 to 1,000 megawatt hours per year and the increased sewage generation would be no more than 1.59 million liters per year (420,000 gal per year), which is less than one percent above existing sitewide levels. At the Nevada Test Site, a central sewage treatment system would have to be constructed for spent nuclear fuel management activities, which would include the foreign research reactor spent nuclear fuel storage facilities. However, all other existing system capacities could manage the estimated increases for materials, utilities, and energy.

4.2.4.2.11 Waste Management

The basic implementation of Management Alternative 1 would only result in minor waste management impacts at the potential foreign research reactor spent nuclear fuel management sites. At all potential management sites the amount of waste generated from foreign research reactor spent nuclear fuel storage is very small when compared to the annual waste projection for each site.

4.2.4.3 Key Cumulative Impacts at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

All of the potential foreign research reactor spent nuclear fuel management sites contain facilities unrelated to foreign research reactor spent nuclear fuel that may continue to operate throughout the foreign research reactor spent nuclear fuel program (approximately 40 years). Impacts from both construction and operation of foreign research reactor spent nuclear fuel facilities would be cumulative with the impacts of existing and planned facilities or actions such as environmental restoration and waste management activities unrelated to foreign research reactor spent nuclear fuel and impacts from the management of DOE's spent nuclear fuel inventory.

This section compares the impacts of the basic implementation of Management Alternative 1 and of the implementation alternatives presented in Section 4.3 to the cumulative impacts at each site. The site-specific cumulative impacts are discussed in more detail in Appendix F.

4.2.4.3.1 Key Cumulative Impacts at the Savannah River Site

Table 4-29 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Savannah River Site, including:

Table 4-29 Key Cumulative Impacts at the Savannah River Site

Environmental Impact Parameter	FRR SNF Receipt and Storage Contribution	FRR SNF Receipt and Chemical Separation Contribution	Current Activities[a]	Other Activities[b]	Cumulative Impact
Occupational and Public Health and Safety:					
• MEI Dose (mrem/yr)	0.00036	0.66	0.25	4.1	5.0
LCF (per year)	1.8×10^{-10}	3.3×10^{-7}	1.25×10^{-7}	0.000002	0.0000025
• Population Dose (person-rem/yr)	0.022	27	9.1	295	331
LCF (per year)	0.000011	0.014	0.0045	0.15	0.17
• Worker Collective Dose (person-rem/yr)	10[c]	21	263	1,418	1700
LCF (per year)	0.004	0.0084	0.10	0.57	0.68
Waste Generation:					
• High-Level (canisters/yr)	0	6.5	(d)	190[e]	190[e]
• Saltstone (m^3/yr)	0	370	(d)	60,000	60,000
• Transuranic (m^3/yr)	0	0	(d)	1,038	1,038
• Mixed/Hazardous (m^3/yr)	0	8	(d)	2,561	2,569
• Low-Level (m^3/yr)	22	5,700	(d)	35,600	41,300

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Based on 1993 site data*

[b] *Other activities include: interim management of nuclear materials, spent nuclear fuel management, Vogtle plant operation, defense waste processing facility, stabilization of plutonium-solutions, site-wide waste management activities, tritium accelerator facility, disposition of surplus HEU, storage and disposition of weapons-usable fissile materials, and the stockpile stewardship and management program activities.*

[c] *The dose is due to the handling of the FRR SNF during receipt and transfer between facility, averaged over 40 years.*

[d] *Included in "other activities"*

[e] *Expected Defense Waste Processing Facility canister production rate (DOE, 1995b).*

- The operation of the Vogtle Electric Generating Plant located approximately 16 km (10 mi) south west of the center of the Savannah River Site.
- The implementation of the preferred alternative in the Management of Nuclear Materials EIS.
- Shipment of aluminum-based spent nuclear fuel to the Savannah River Site for storage and disposal discussed in Appendix C of the Programmatic SNF & INEL Final EIS.
- Completion of the construction and operation of the Defense Waste Processing Facility.
- Processing of F-Canyon plutonium solutions to metal.
- Treatment and minimization of radioactive and hazardous wastes at the site as identified in the Savannah River Site Waste Management Final EIS.
- Construction of an accelerator for tritium production at the Savannah River Site, along with associated support facilities.
- Disposition of Surplus Highly Enriched Uranium at the site.
- Storage and Disposition of Weapons-Usable Fissile Materials.

- Stockpile Stewardship and Management Program.
- Current Savannah River Site projects (based on 1993 data).

Table 4-29 also shows the impacts of receipt and near-term chemical separation at the Savannah River Site, from Implementation Alternative 6 of Management Alternative 1 in Section 4.3.6. These impacts are sufficiently distinct from those of the other alternatives that they are presented separately. These impacts would occur only while the chemical separation facilities are operating.

The results in Table 4-29 show that the contribution of foreign research reactor spent nuclear fuel to the cumulative impacts at the Savannah River Site would be minimal.

4.2.4.3.2 Key Cumulative Impacts at the Idaho National Engineering Laboratory

Table 4-30 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Idaho National Engineering Laboratory, including the proposed construction and operation of an accelerator facility for tritium production (along with associated support facilities), the management of DOE-owned spent nuclear fuel discussed in Appendix B of the Programmatic SNF&INEL Final EIS, and the storage and disposition of weapons-usable fissile materials at the Idaho National Engineering Laboratory site.

Table 4-30 Key Cumulative Impacts at the Idaho National Engineering Laboratory

Environmental Impact Parameter	*FRR SNF Receipt and Storage Contribution*	*FRR SNF Receipt and Chemical Separation Contribution*	*Current Activities*[a]	*Other Activities*[a]	*Cumulative Impact*
Occupational and Public Health and Safety:					
• MEI Dose (mrem/yr)	0.00056	0.048	0.056	0.0057	0.11
LCF (per year)	2.8×10^{-10}	2.4×10^{-8}	2.8×10^{-8}	2.8×10^{-9}	5.5×10^{-8}
• Population Dose (person-rem/yr)	0.0045	0.39	0.34	32	33
LCF (per year)	2.3×10^{-6}	0.00020	0.00017	0.016	0.016
• Worker Collective Dose (person-rem/yr)	10[b]	18	30	344	392
LCF (per year)	0.004	0.0072	0.012	0.137	0.16
Waste Generation:					
• High-Level (canisters/yr)	0	7.5	0	327[c]	327[c]
• Grout (m^3/yr)	0	167	0	875[d]	875[d]
• Transuranic (m^3/yr)	0	0	712	46	758
• Mixed/Hazardous (m^3/yr)	0	8	243	8	259
• Low-Level (m^3/yr)	22	5,700	4,795	2,800	13,300

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of a tritium accelerator facility, and the disposition of weapons-usable fissile materials.*

[b] *The dose is due to the handling of FRR SNF during receipt and transfer, averaged over 40 years.*

[c] *Assumed canister production rate (DOE, 1995b).*

[d] *Design capacity of the proposed Waste Immobilization Facility, which is not funded.*

Table 4-30 also shows the impacts of receipt and near-term chemical separation at the Idaho National Engineering Laboratory, from Implementation Alternative 6 of Management Alternative 1 in Section 4.3.6. These impacts are sufficiently distinct from those of the other alternatives that they are presented separately. These impacts would occur only while the chemical separation facilities are operating.

The results in Table 4-30 show that the contribution of foreign research reactor spent nuclear fuel management to the cumulative impacts at the Idaho National Engineering Laboratory would be minimal.

4.2.4.3.3 Key Cumulative Impacts at the Hanford Site

Table 4-31 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Hanford Site, including those discussed in the Programmatic SNF&INEL Final EIS, the Management of Spent Nuclear Fuel from the K Basins Draft EIS, and the Safe Interim Storage of Hanford Tank Wastes Final EIS.

Table 4-31 Key Cumulative Impacts at the Hanford Site

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Occupational and Public Health and Safety:			
• MEI Dose (mrem/yr)	0.00025	0.0036	0.0036
LCF (per year)	1.3×10^{-10}	1.5×10^{-9}	1.5×10^{-9}
• Population Dose (person-rem/yr)	0.015	0.22	0.235
LCF (per year)	0.0000075	0.00011	0.00011
• Worker Collective Dose (person-rem/yr)	8.9[b]	116.5	125.4
LCF (per year)	0.0035	0.0466	0.05
Waste Generation:			
• High-Level (canisters/yr)	0	320[c]	320[c]
• Transuranic (m^3/yr)	0	240	240
• Mixed/Hazardous (m^3/yr)	0	402	402
• Low-Level (m^3/yr)	22	33,310	33,332

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of a Laser Interferometer Gravitational-Wave Observatory, decommissioning of unused facilities, site restoration activities, interim storage and tank wastes, management of spent nuclear fuel from the K basins, and current activities.*

[b] *The dose is due to the handling of FRR SNF during receipt, averaged over 30 years.*

[c] *Assumed canister production rate (DOE, 1995b).*

The results in Table 4-31 show that the contribution from management of foreign research reactor spent nuclear fuel to the cumulative impacts at the Hanford Site would be minimal.

4.2.4.3.4 Key Cumulative Impacts at the Oak Ridge Reservation

Table 4-32 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Oak Ridge Reservation, including those discussed in the programmatic SNF&INEL Final EIS, the Tritium Supply and Recycling Final EIS, and the Disposition of Surplus Highly Enriched Uranium Draft EIS. Other activities considered for the Oak Ridge Reservation which could affect the site environment have not been determined sufficiently at this time to allow impact evaluation. They include activities associated with the waste management at the site, storage and disposition of weapons-usable fissile materials, and stockpile stewardship and management program.

Table 4-32 Key Cumulative Impacts at the Oak Ridge Reservation

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Occupational and Public Health and Safety:			
• MEI Dose (mrem/yr)	0.09	15.5	15.6
LCF (per year)	4.5×10^{-8}	0.0000077	0.0000078
• Population Dose (person-rem/yr)	0.085	94.5	94.6
LCF (per year)	0.000043	0.047	0.047
• Worker Collective Dose (person-rem/yr)	8.9[b]	261.3	270.2
LCF (per year)	0.0036	0.104	0.108
Waste Generation:			
• High-Level (canisters/yr)	0	0	0
• Transuranic (m^3/yr)	0	16	16
• Mixed/Hazardous (m^3/yr)	0	119,411	119,411
• Low-Level (m^3/yr)	22	34,989	35,011

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of the Expended Core Facility, the construction and operation of the Advanced Neutron Source Facility, construction and operation of a Tritium production facility, and surplus highly-enriched uranium management activities at the site.*

[b] *The dose is due to the handling of FRR SNF during receipt, averaged over 30 years.*

The results in Table 4-32 show that the contribution from storage of foreign research reactor spent nuclear fuel to the cumulative impacts at the Oak Ridge Reservation would be minimal.

4.2.4.3.5 Key Cumulative Impacts at the Nevada Test Site

Table 4-33 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Nevada Test Site, including those discussed in the Programmatic SNF&INEL Final EIS and the Tritium Supply and Recycling Final EIS. The Programmatic SNF&INEL Final EIS includes the quantitative impacts from a proposed Expended Core Facility at the Site. The Nevada Test Site is also considered in the storage and disposition of weapons-usable fissile materials program which could affect the site environment. The impacts from this program have not been determined sufficiently at this time to allow impact evaluation.

The results in Table 4-33 show that the contribution from storage of foreign research reactor spent nuclear fuel to the cumulative impacts at the Nevada Test Site would be minimal.

4.2.4.4 Waste Minimization and Mitigation Measures at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

Although environmental impacts at the potential foreign research reactor spent nuclear fuel management sites would be minimal in all environmental media and mitigation measures would not be necessary, the sites would implement measures in some areas to minimize impacts. Mitigation measures would be taken in the areas of pollution control, socioeconomics, cultural resources, air and water resources, occupational and public health and safety, and accident prevention. Appendix F provides details on these issues.

Table 4-33 Key Cumulative Impacts at the Nevada Test Site

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Occupational and Public Health and Safety:			
• MEI Dose (mrem/yr)	0.00076	0.31	0.31
LCF (per year)	3.8×10^{-10}	1.55×10^{-7}	1.55×10^{-7}
• Population Dose (person-rem/yr)	0.00093	0.095	0.095
LCF (per year)	4.7×10^{-7}	0.00047	0.000047
• Worker Collective Dose (person-rem/yr)	8.9[b]	81	89.9
LCF (per year)	0.0036	0.032	0.035
Waste Generation:			
• High-Level (canisters/yr)	0	0	0
• Transuranic (m^3/yr)	0	16	16
• Mixed/Hazardous (m^3/yr)	0	252	252
• Low-Level (m^3/yr)	22	44,578	44,600

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include existing activities, DOE-owned spent fuel management activities, construction and operation of an Expended Core Facility, and construction and operation of a tritium production facility.*

[b] *The dose is due to the handling of foreign research reactor spent nuclear fuel during receipt, averaged over 30 years.*

4.2.4.5 Environmental Justice at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

Under incident-free foreign research reactor spent nuclear fuel management site activities associated with receipt and storage of the spent nuclear fuel, the dominant radiological impacts would be the exposures received by the site workers in the immediate vicinity of the spent nuclear fuel container. These individuals are principally those working within the spent nuclear fuel storage facility. As discussed in Section 4.2.4.1, under incident-free operating conditions, no radiological fatalities would be expected among radiation workers or the general public.

Section 4.2.4.1 also discusses radiological effects due to accidents for both wet storage and dry storage. As shown in Tables 4-24 through 4-28, the dominant radiological risks due to accidents are estimated to occur during breach of a spent nuclear fuel assembly. No LCF are expected to result from the basic implementation of Management Alternative 1.

Appendix A describes minority populations and low-income households residing near candidate management sites. Table 4-34 summarizes this description. Calculations for incident-free and accident conditions demonstrate that for the general population the impacts would be very low. Minority or low-income populations living near the potential management sites would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. They would be subject to very low impacts, as would the general population.

Table 4-34 Summary Description of Minority Populations and Low-Income Households Residing Within 80 km (50 mi) of Candidate Management Sites

Candidate Management Site	*Total Population*	*Minority Population*	*Total Households*	*Low-Income Households*
Savannah River Site	566,823	214,016	197,937	82,930
Idaho National Engineering Laboratory	176,311	15,449	55,109	22,452
Hanford Site	383,934	95,042	136,496	57,667
Oak Ridge Reservation	863,758	53,185	335,589	147,537
Nevada Test Site	12,421	2,005	4,194	2,024

Characterization of the number and location of minority and low-income populations is dependent on how these populations are defined and what assumptions are used in conducting the analysis. As discussed in Appendix A, at the time this Final EIS and the Programmatic SNF&INEL Final EIS were prepared, the Federal Interagency Working Group on Environmental Justice had not issued final guidance on the definitions of minority and low-income populations, or the approach to be used in analyzing environmental justice, as directed by the Executive Order. Final internal DOE guidance on environmental justice has also not been adopted. As a result, both the definitions and assumptions used by and within agencies for conducting environmental justice analyses can vary, and the resulting demographic results can differ on a case-by-case basis. For example, this Final EIS and the Programmatic SNF&INEL Final EIS present demographic characterizations derived from the same United States Census Bureau data base, but these documents used different definitions and assumptions. Several of the same candidate interim spent nuclear fuel management sites were evaluated in both documents. As discussed in Appendix A, variations in these definitions and assumptions led to differences in the characterization of minority and low-income populations surrounding these potential spent nuclear fuel management sites. Nevertheless, although the characterizations differ, the radiological impacts resulting from the proposed action under all alternatives present very low risk to the population as a whole. Therefore, no disproportionately high and adverse effects would be expected for any particular segment of the population, including minority and low-income populations, regardless of which set of definitions and assumptions were applied.

Implementation of the proposed action would have extremely low nonradiological effects on the environment at interim management sites, including the social and economic status of the general population, minority populations, and the low-income population surrounding interim management sites. Economic benefits that would result from increased cargo handling, transportation, and storage at interim management sites would be extremely small for the general population or any particular segment of the population residing near interim management sites.

4.2.4.6 Mitigation Measures at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

Based on the analyses of the environmental consequences for each potential foreign research reactor spent nuclear fuel management site included in Section F.4 of Appendix F, no mitigation measures would be necessary since all potential environmental impacts are substantially below acceptable levels or promulgated standards. However, each potential site would follow operation practices that would minimize the impacts in such areas as pollution prevention, cultural and ecological resources, ground and surface water quality, air quality, noise, traffic, operational and public health and safety, and accident prevention and mitigation. Descriptions of these practices are included in Appendix F, under Mitigation Measures for each site.

4.2.5 Short-Term Uses and Long-Term Productivity

Short-term impacts would be those associated with construction and operation of the storage facilities. No land would be used for the marine or ground transportation of foreign research reactor spent nuclear fuel. The use of land at the potential foreign research reactor spent nuclear fuel management sites would be in conformity with the land use policy of each site. The construction of new storage facility would lead to the loss of small acreage of terrestrial habitat. After adoption of an overall strategy for interim storage of all DOE-owned spent nuclear fuel (including spent fuel from foreign research reactors), some of the areas currently used for interim storage of spent nuclear fuel may be released for other productive uses

(DOE, 1995c). Ecological resources would be directly affected at the area of construction by land clearing. These resources would be limited to small mammals, reptiles, and songbirds. Given the small area that would be used, the overall effect would be of limited impacts on local populations and resources.

4.2.6 Irreversible and Irretrievable Commitments of Resources

The only irreversible use of resources during the marine and ground transportation of foreign research reactor spent nuclear fuel would be the use of petroleum fuel. Irreversible and irretrievable commitment to resources associated with management site activities are discussed below.

4.2.6.1 Management Site Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of foreign research reactor spent nuclear fuel management site facilities would involve materials that could not be recovered or recycled, or resources that would be consumed or reduced to unrecoverable forms, including electrical energy, fuel, construction materials, and miscellaneous chemicals. Some construction materials are recyclable. Some of the resources would be irretrievable because of the nature of the commitment or the cost of reclamation. For example, human resources used for the construction and operation of the potential foreign research reactor spent nuclear fuel storage facilities would be irretrievably lost since these resources would be unavailable for use in other work activity areas. On the whole, foreign research reactor spent nuclear fuel management would not be particularly resource-intensive. The quantities of irreversible and irretrievable resources for each site are included in Appendix F, Section F.4.

4.2.6.2 Energy Resources

Under the basic implementation of Management Alternative 1, about 4.6 metric tons (5.1 tons) of highly-enriched uranium would be accepted into the United States. The energy content of this uranium would be equal to about 1.5 million megawatt-days or over 20 million barrels of No. 2 fuel oil if the conversion efficiency were 100 percent.

4.2.7 Impacts of Ultimate Disposition

Ultimate disposition of DOE's spent nuclear fuel, including foreign research reactor spent nuclear fuel, is a high priority. For planning purposes, DOE has determined that its spent nuclear fuel that is not otherwise managed (e.g., chemically separated, with the high-level waste being converted into a vitrified glass for repository disposal) is authorized for disposal in a geologic repository. Decisions regarding the actual disposition of DOE's spent nuclear fuel will follow appropriate review under the National Environmental Policy Act (NEPA).

It is possible that the foreign research reactor spent nuclear fuel could be accepted intact in a geologic repository. If DOE determines that geologic disposal of intact foreign research reactor spent nuclear fuel is possible, then there would be no onsite impacts beyond those associated with storage and packaging of the foreign research reactor spent nuclear fuel.

It is also possible that some form of processing could be necessary to convert the foreign research reactor spent nuclear fuel into a more stable form prior to its ultimate disposal. This processing could be a near-term new treatment technology, conventional chemical separation, or a new treatment technology that is implemented after an interim period of storage. DOE expects that any new treatment technology would produce no greater impacts than historical chemical separation activities. Therefore, the impacts of near-term treatment of the foreign research reactor spent nuclear fuel would be expected to be no greater than the impacts of chemically separating the same material as discussed in Section 4.3.6. If a new treatment technology is implemented after an interim period of storage and technology development, DOE expects that it would provide a substantial improvement over conventional chemical separation.

When disposal space is available, DOE would transport the intact or processed foreign research reactor spent nuclear fuel to a repository. This transportation would produce impacts similar to the ground transportation impacts discussed in Section 4.4.2.3. Handling and emplacement in the repository would produce impacts similar to those due to handling the spent nuclear fuel or processed waste at the DOE site because similar equipment and procedures would be used and the same regulatory limits on radiation doses would apply.

Yucca Mountain is the candidate site for a geologic repository for both spent nuclear fuel and high-level waste. Under the Nuclear Waste Policy Act, Congress found that a national problem had been created by the accumulation of spent nuclear fuel from commercial reactors and the accumulation of high-level waste. The Nuclear Waste Policy Act assigned to DOE the responsibility for managing the disposal of this spent nuclear fuel and high-level waste, specified the siting process, and authorized the construction of one geologic repository. Under the Nuclear Waste Policy Act Amendments Act of 1987, the process for selecting this repository was streamlined, and the Yucca Mountain site in Nevada was selected as the candidate site for a geologic repository.

Because the environmental documentation process for geologic disposal was established by the Nuclear Waste Policy Act, this EIS does not analyze environmental impacts of disposal at Yucca Mountain or alternative locations. After emplacement in a geologic repository, however, DOE expects there would be no more impacts to workers, the public, or the environment because the radioactive material would be effectively isolated.

In the event that a geologic repository were to be delayed, DOE assumed for purposes of this analysis that it would continue to manage the foreign research reactor spent nuclear fuel, or the high-level radioactive waste resulting from the chemical separation or other processing of such spent nuclear fuel, at the management sites until a geologic repository becomes available. The risk associated with this continued management is low and would not exceed the annual risk discussed in Section 4.2.4.1.

4.2.8 Summary of the Impacts of the Basic Implementation of Management Alternative 1

The principal impacts under the basic implementation of Management Alternative 1 would be occupational and public health and safety impacts. These are presented in Table 4-35 in terms of the risk of death due to cancer for each segment of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-35 shows that the greatest radiological risks would occur during ground transport or site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel produces a dose rate equal to the regulatory limit, (2) truck shipments expose people at highway rest stops for times about equal to the actual driving times, and (3) one individual at the DOE site receives the maximum dose allowed by DOE regulation (5,000 mrem) every year.

Table 4-35 Maximum Estimated Radiological Health Impacts of the Basic Implementation of Management Alternative 1

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	Population	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 0.000029	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2×10^{-10}	0.000029	---
Ground Transport			
Incident-Free	0.00052	0.22	0.071
Accidents	1.4×10^{-11}	0.00028	---
Site Activities			
Incident-Free	0.026	0.00027	0.21
Accidents	0.000010	0.11	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000010	----	----
Total Population Risk			
Incident-Free	----	0.22	0.33
Accidents	----	0.11	----

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 1.4×10^{-7} LCF.

The highest estimated accident MEI risk is 0.000010 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be one in one hundred thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-35, the total incident-free population risk would be 0.22 LCF for the potentially exposed public, while the corresponding risk would be 0.33 LCF for workers. Thus, there would be an estimated 22 percent chance of incurring one additional LCF among the exposed general public, and a 33 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-35. DOE and the Department of State estimate there could be about a 14 percent chance that a truck driver or member of the public could die in a traffic accident associated with the basic implementation of Management Alternative 1. This death would result from the traffic accident trauma and would be unrelated to the radioactive nature of the cargo.

4.3 Implementation Alternatives of Management Alternative 1

As discussed in Chapter 2, a policy of managing foreign research reactor spent nuclear fuel in the United States could be implemented by various means. These variations on the basic implementation of Management Alternative 1 of the proposed action have been grouped into seven implementation alternatives. This section discusses their policy considerations and environmental impacts. For convenience, the seven implementation alternatives are listed briefly below:

1. Acceptance of amounts of material different from the amount in the basic implementation of Management Alternative 1,

2. Acceptance of foreign research reactor spent nuclear fuel for periods of time different from the period of time in the basic implementation of Management Alternative 1,

3. Implementation through financial arrangements different from those identified in the basic implementation of Management Alternative 1,

4. Taking title to the foreign research reactor spent nuclear fuel at locations different from the location in the basic implementation of Management Alternative 1,

5. Use of wet storage technology for the interim period instead of dry storage technology as in the basic implementation of Management Alternative 1,

6. Near term conventional chemical separation of the foreign research reactor spent nuclear fuel instead of interim storage as in the basic implementation of Management Alternative 1, and

7. Development and use of a new processing technology instead of interim storage as in the basic implementation of Management Alternative 1.

4.3.1 Implementation Alternative 1: Alternative Amounts of Spent Nuclear Fuel to be Accepted

DOE and the Department of State have evaluated the policy considerations and environmental impacts for different amounts of spent nuclear fuel and target materials under this implementation alternative.

4.3.1.1 Implementation Subalternative 1a: Accept Foreign Research Reactor Spent Nuclear Fuel Only From Developing Nations

Policy Considerations

Under this implementation subalternative, up to 1.9 MTHM and about 5,000 elements of foreign research reactor spent nuclear fuel would be accepted into the United States from developing nations (defined by the World Bank as nations with other-than-high-income economies). Up to about 238 kg (525 lb) of HEU would be removed from international commerce. By excluding developed countries, which generally share our nuclear weapons nonproliferation goals, but do not necessarily share our belief in the necessity for removing HEU from use in civil programs, this subalternative would have adverse consequences for U.S. nuclear weapons nonproliferation policy.

Because the United States has been unable to accept shipments of HEU spent nuclear fuel since 1988, several foreign research reactor operators have run out of storage capacity or face safety and regulatory problems associated with the presence of spent nuclear fuel at their sites. If the United States is unable to accept any near term shipments of spent nuclear fuel from developed countries, some reactor operators will be forced to either shut down their reactors or ship their spent nuclear fuel for reprocessing to the United Kingdom Atomic Energy Authority facility in Dounreay, United Kingdom, which is the only facility currently able and willing to reprocess foreign research reactor spent nuclear fuel. Operators in Belgium and Germany have already sent spent nuclear fuel elements to Dounreay for reprocessing. Since neither Dounreay nor any other facility is currently accepting aluminum-based research reactor spent nuclear fuel containing LEU for reprocessing, the only way a reactor operator can use reprocessing to control his spent nuclear fuel inventory is by using HEU for fuel. This could lead reactor operators to delay or cancel plans to convert to LEU, or, in some cases, to reconvert from LEU to HEU fuels.

The net result of reduced reliance on the United States is that foreign research reactor operators would be compelled to withdraw from the Reduced Enrichment for Research and Test Reactors (RERTR) program and continue operations on the HEU fuel cycle, with its lower costs and enhanced performance. Since the United States is barred from exporting HEU to virtually all foreign research reactors under the Energy Policy Act of 1992, operators would be forced to seek alternative suppliers of HEU, such as Russia and China. This could lead to renewed international commerce in weapons-usable HEU and undermine the U.S. nuclear weapons nonproliferation policy goal of seeking to minimize the civil use of HEU. Further, those countries that participated in the RERTR program considered U.S. acceptance of their spent nuclear fuel as a condition for incurring the substantial costs and technical difficulties of converting to LEU fuels. Failure to accept their spent nuclear fuel would jeopardize the nuclear weapons nonproliferation goals of the RERTR program and the reputation of the United States as a reliable partner in the conduct of international nuclear materials management.

There is another way this subalternative could undercut the RERTR program. The developing countries generally assess their technical capabilities by comparing themselves with the developed states of North America, Western Europe, and Japan. As noted above, one probable result of this subalternative is that more developed states will continue to use HEU-fueled research reactors, due to difficulty in reprocessing LEU spent nuclear fuel. If that happens, developing countries are likely to regard use of HEU-fueled reactors as more advanced and prestigious than LEU-fueled reactors, increasing the demand for such reactors as well as for HEU itself. Again, this would encourage increased stockpiles of HEU in various developed and developing countries, contrary to U.S. nuclear weapons nonproliferation policy.

If some countries are forced to shut down their reactors and thereby forego the medical and scientific benefits of these reactors, such a situation may lead to criticism that the United States is not a dependable nuclear partner. Some countries, including those in the developing world that have characterized the Treaty on the Non-Proliferation of Nuclear Weapons as a discriminatory bargain between the nuclear "haves" and the nonnuclear "have-nots," may be inclined to accuse the United States, fairly or unfairly, of having failed to comply with its Article IV Treaty pledge to facilitate "the fullest possible exchange of equipment, materials and scientific and technological information for the peaceful uses of nuclear energy." Actions that foster such negative perceptions would undoubtedly complicate the conferences which are scheduled to monitor compliance with the Non-Proliferation Treaty, and may complicate United States diplomatic efforts to attain other arms control and nuclear weapons nonproliferation objectives.

Marine Transport Impacts

Impacts of Incident-Free Marine Transport

The impacts of incident-free marine transportation were analyzed in the same manner as the basic implementation of Management Alternative 1. The incident-free transportation of spent nuclear fuel was estimated to result in total LCF that ranged from 0.008 to 0.009 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the ships' crews.

The range of impacts result from the analysis of shipment of the spent nuclear fuel on regularly scheduled commercial breakbulk vessels and on chartered container vessels and would be the same as for vessels analyzed in the evaluation of the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the difference between the two estimates is a result of the shorter vessel journey time for chartered vessels due to the intermediate port stops associated with the regularly scheduled commercial transport of the spent nuclear fuel.

Impacts of Accidents During Marine Transport

The consequences of the at-sea accidents for Implementation Subalternative 1a are no different than the consequences of at-sea accidents associated with the basic implementation of Management Alternative 1. For an accident involving the loss of a transportation cask in coastal waters, the maximum exposure to an individual is estimated to be 114 mrem per year. Due to the reduced number of cask shipments, the likelihood of such an accident would be reduced. Under this subalternative, 23 percent of the total number of cask shipments required under the basic implementation of Management Alternative 1 would be needed. The highest estimated risks due to an accident during marine transport would therefore be 0.00004 mrem per year peak dose to a human from the loss of a damaged cask in the deep ocean. This corresponds to an MEI risk of 1×10^{-10} LCF. This individual would have a chance of less than one in a billion of incurring an LCF due to an accident during marine transport.

Port Activity Impacts

Impacts of Incident-Free Port Activities

In the analysis of the basic implementation of Management Alternative 1, the radiological impact of port activities was estimated on a per-shipment basis. Implementation Subalternative 1a, accepting spent nuclear fuel from developing nations only, results in 23 percent of the total number of cask shipments that are required under the basic implementation of Management Alternative 1. The incident-free impacts of the port activities would be proportionally reduced. The estimated number of LCF associated with this subalternative range from 0.0008 to 0.003. As in the marine incident-free analysis, this range of impacts is the result of the analysis of two modes of spent nuclear fuel shipment, regularly scheduled commercial breakbulk vessels and chartered container vessels.

Impacts of Accidents During Port Activities

Port accident risks were calculated based on the per-shipment risks determined in the analysis of the basic implementation of Management Alternative 1. The analysis examined the impact of using a wide range of ports of entry based on the population around the port city, from high density population ports such as Elizabeth, NJ, to low-density ports such as the MOTSU terminal in North Carolina. The analysis also considered the impact of chartered shipments (no intermediate port stops before the vessel reaches the spent nuclear fuel port of entry) versus regularly scheduled commercial shipments with up to two intermediate ports of call before the spent nuclear fuel port of entry. The port accident risks over the entire program are estimated to range from 5 x 10^{-8} to 0.000004 LCF from radiation. The range of fatality estimates is due to both the differences in port city populations and the number of intermediate port stops.

The consequences of the maximum foreseeable port accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced number of shipments, so the MEI risk is reduced to 5 x 10^{-11} LCF.

Ground Transport Impacts

Impacts of Incident-Free Ground Transport

Radiological impacts of incident-free ground transportation were analyzed in the same manner for Implementation Subalternative 1a as for the basic implementation of Management Alternative 1. The results are presented in Figures 4-6 through 4-9. Incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.002 to 0.06 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates is caused by two factors: the option of using truck or rail to transport spent nuclear fuel and combinations of Phase 1 and Phase 2 potential foreign research reactor spent nuclear fuel management sites that created varying shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.001 to 0.015. The estimated number of radiation-related LCF for the general population ranged from 0.0006 to 0.045, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0002 to 0.01.

Impacts of Accidents During Ground Transport

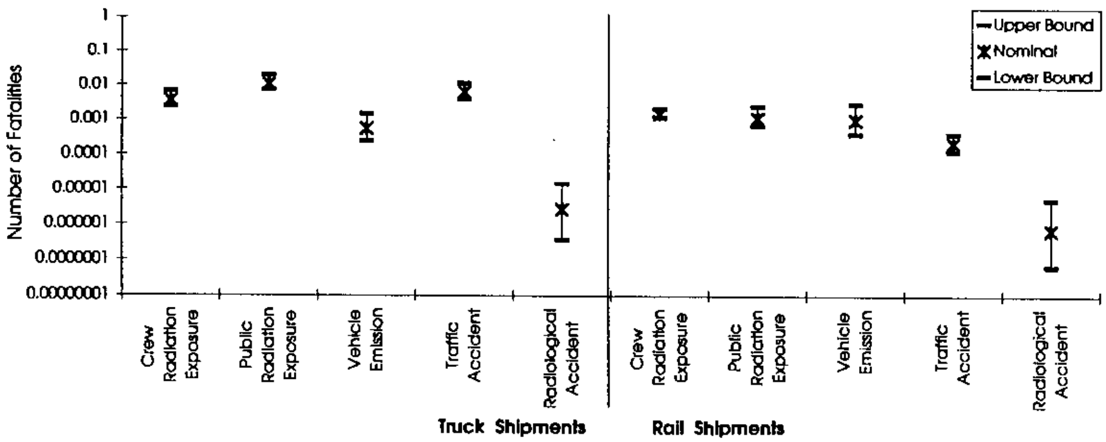
The transportation accident population risks over the entire program are estimated to range from 0.0000001 to 0.00006 LCF from radiation and from 0.0001 to 0.028 traffic fatalities, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation.

The maximum foreseeable offsite transportation accident is identical to that for the basic implementation of Management Alternative 1. The risk is reduced to 2.7 x 10^{-12} LCF due to the reduced amount of ground transport.

Management Site Impacts

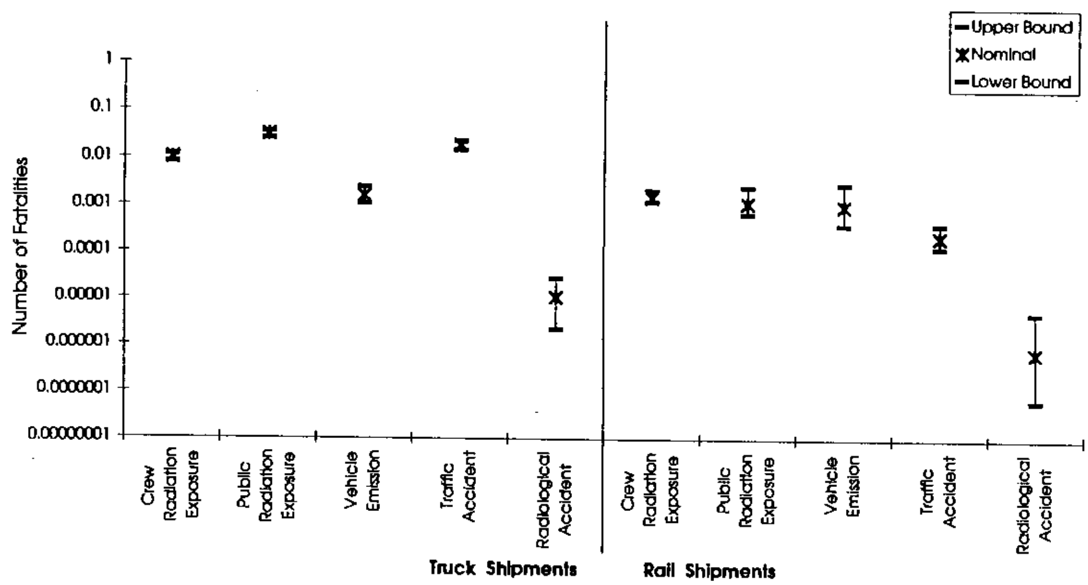
Impacts of Incident-Free Management Site Activities

Impacts of incident-free site activities from Implementation Subalternative 1a are covered by the impacts from the basic implementation of Management Alternative 1. The maximally exposed worker radiation dose depends upon the duration of the receipts, not the amount of foreign research reactor spent nuclear fuel involved, and the duration in this subalternative is identical to the basic implementation of Management Alternative 1 (13 years). Thus, the maximally exposed worker dose is conservatively



- *Same values apply to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives*
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-6 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1a and the Programmatic SNF&INEL Final EIS Decentralization Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-7 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1a and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative

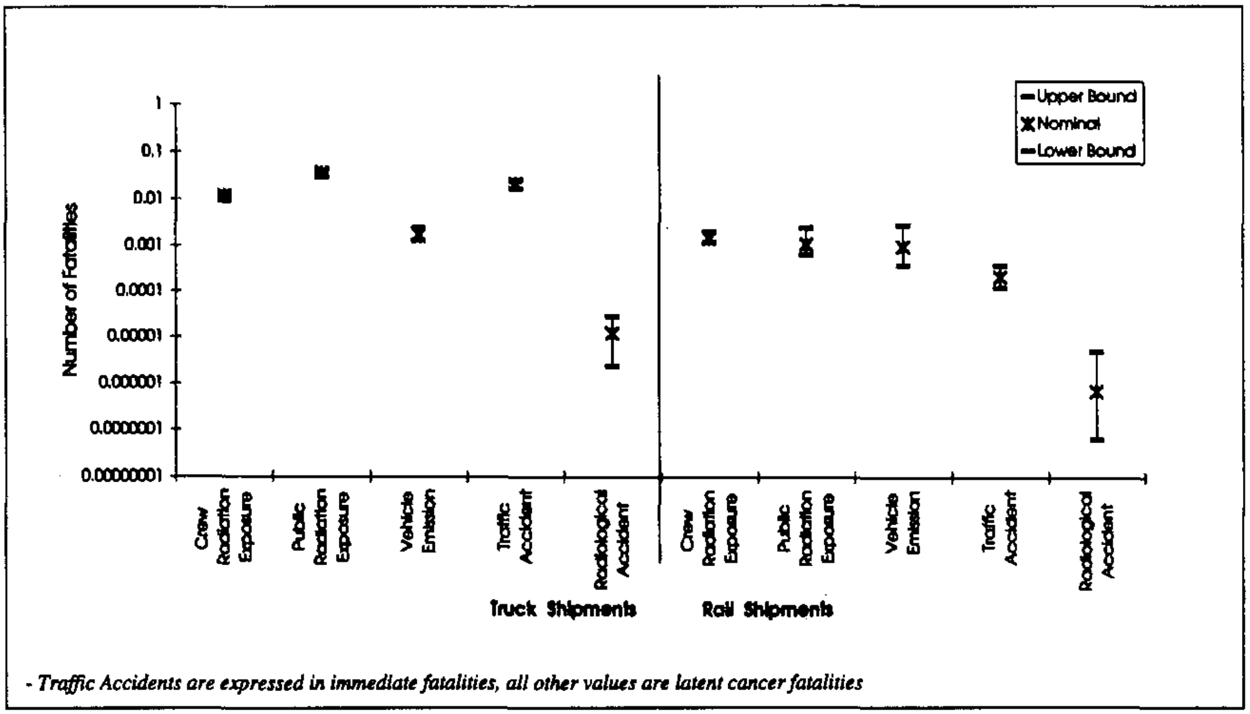


Figure 4-8 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1a and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative

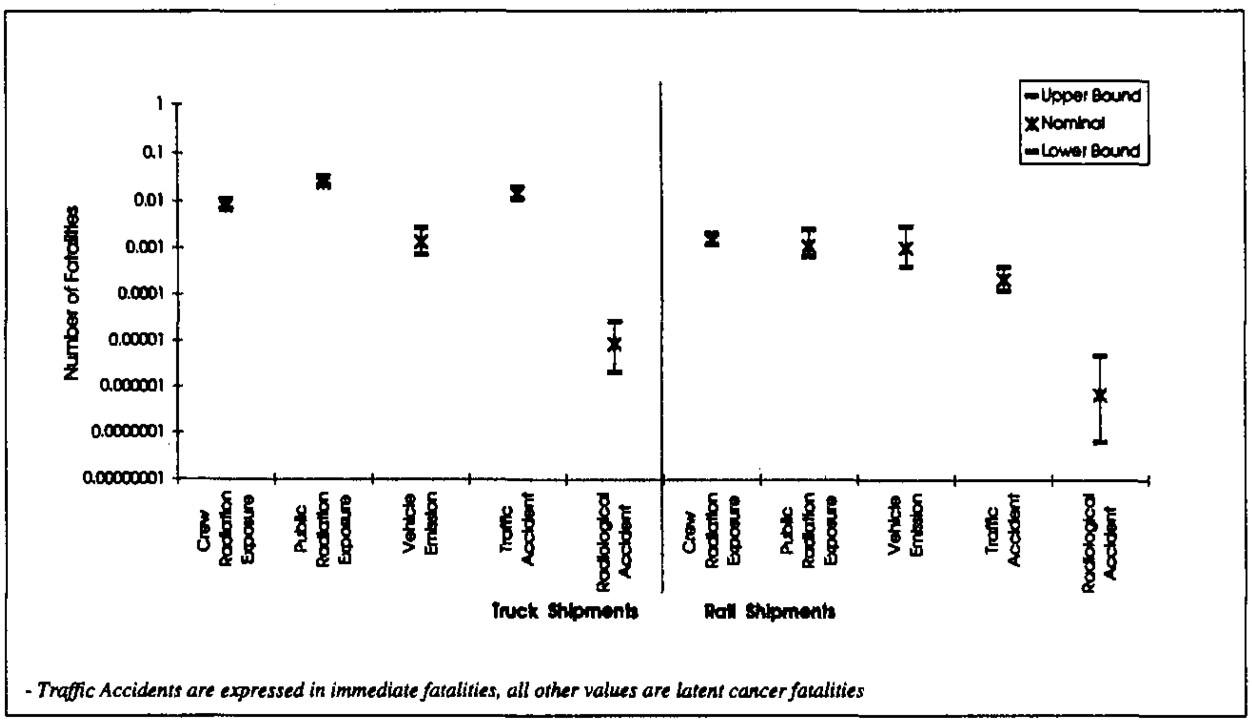


Figure 4-9 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1a and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

assumed to be the same as in the basic implementation of Management Alternative 1. This would produce the maximally exposed worker risk identical to that in the basic implementation of Management Alternative 1 of 0.026 LCF.

The amount of foreign research reactor spent nuclear fuel that would be received and managed is 5,000 elements or approximately 22 percent of the number of elements in the basic implementation of Management Alternative 1. Thus, it is expected that the worker population risks at each management site would be approximately 22 percent of those calculated for the basic implementation of Management Alternative 1. The highest estimate of this risk under the basic implementation of Management Alternative 1 is 0.21 LCF, so the corresponding risk for this subalternative is 0.05, LCF, which is much less than one LCF.

Similarly, some of the incident-free public risk depends on the amount of foreign research reactor spent nuclear fuel involved and some depends on the duration of each activity. The risk that accrues during receipt and handling can be scaled down by the factor of 22 percent, while the risk that accrues during storage is dependent only on the duration of the storage. The highest estimated incident-free MEI risk in the basic implementation of Management Alternative 1 (1.4×10^{-7} LCF) is due to receipt and handling, so it is reduced by a factor of 22 percent to yield the corresponding risk of 3.1×10^{-8} LCF for this subalternative.

The highest estimated incident-free public population risk in Phase 1 of the basic implementation of Management Alternative 1 (0.00014 LCF) is due to storage, so it is not reduced in this subalternative. The corresponding Phase 2 risk (0.00013 LCF) is due to receipt and handling, so this component of the risk is reduced to 0.000029 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.00017 LCF.

Impacts of Accidents Onsite

The highest estimated MEI risk due to accidents in the basic implementation of Management Alternative 1 (0.0000034 LCF) is due to an accidental criticality in RBOF. This MEI risk is greater than any of the potential Phase 2 MEI risks, when those due to receipt/handling are reduced by the factor of 22 percent. Thus, the highest MEI risk due to accidents is 0.0000034 LCF.

The highest estimated population risk due to Phase 1 accidents in the basic implementation of Management Alternative 1 (0.096 LCF) is due to an accidental criticality in RBOF. The same facility could be used for the same period of time in this subalternative, so this component of the risk is unchanged. The corresponding Phase 2 risk (0.013 LCF) is due to receipt and handling, so it is reduced by the factor of 22 percent to 0.0029 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.099 LCF.

Summary of the Impacts of Implementation Subalternative 1a

The principal impacts under this subalternative would be occupational and public health and safety impacts. These are presented in Table 4-36 in terms of the risk of death due to cancer during each of the four segments of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-36 Maximum Estimated Radiological Health Impacts of Implementation Subalternative 1a (Developing Nations Only)

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.009
Accidents	1×10^{-10}	much less than 0.000004	---
Port Activities			
Incident-Free	0.00052	0	0.003
Accidents	5×10^{-11}	0.000004	---
Ground Transport			
Incident-Free	0.00052	0.045	0.015
Accidents	2.7×10^{-12}	0.00006	---
Site Activities			
Incident-Free	0.026	0.00017	0.05
Accidents	0.0000034	0.099	---
Highest Individual Risk			
Incident-Free	0.026	----	----
Accidents	0.0000034	----	---
Total Population Risk			
Incident-Free	----	0.045	0.077
Accidents	----	0.099	----

Table 4-36 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel produces a dose rate equal to the regulatory limit, (2) truck shipments exposes people at highway rest stops for times about equal to the actual driving times, and (3) one individual at the DOE management site receives the maximum dose allowed by DOE regulation (5,000 mrem) every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated indicent-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 3.1×10^{-8} LCF.

The highest estimated accident MEI risk is 0.0000026 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in one hundred thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-36, the total incident-free population risk would be 0.045 LCF for the potentially exposed public, and the corresponding risk would be 0.077 LCF for workers. Thus, there would be less than a five percent chance of incurring one additional LCF among the general public, and a 7.7 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-36. There is about a three percent chance that a truck driver or member of the public could die in a traffic accident associated with this subalternative. This death would result from the traffic accident trauma and would be unrelated to the radioactive nature of the cargo.

4.3.1.2 Implementation Subalternative 1b: Accept Only Foreign Research Reactor Spent Nuclear Fuel that Contains HEU

Policy Considerations

Under this implementation subalternative, up to about 4.6 MTHM and 11,200 elements of foreign research reactor spent nuclear fuel would be accepted into the United States. All of this foreign research reactor spent nuclear fuel would contain HEU that was enriched in the United States.

Although this implementation subalternative would remove up to about 4.6 metric tons (5.1 tons) of HEU from international commerce, it almost certainly would result in the end of the RERTR program. As discussed in Chapter 1, the foreign research reactor operators have stated that they would not participate in the RERTR program unless the United States accepts their spent nuclear fuel, including LEU spent nuclear fuel. Otherwise, many research reactor operators would be likely to insist on using HEU fuel in their reactors in the future, which would increase international commerce in HEU. The most likely suppliers of this HEU would be Russia and China. DOE and the Department of State believe that in the long run, this subalternative would be contrary to the broader U.S. policy of nuclear weapons nonproliferation. Therefore, this subalternative is not analyzed in detail for environmental impacts in this EIS.

Summary of the Impacts of Implementation Subalternative 1b

Since the number of elements in this subalternative is about half the number of elements in the basic implementation of Management Alternative 1, the impacts would be roughly half of those calculated for the basic implementation of Management Alternative 1 (see Section 4.2.8).

4.3.1.3. Implementation Subalternative 1c: Accept Target Material in Addition to Foreign Research Reactor Spent Nuclear Fuel

Policy Considerations

This implementation subalternative would entail the shipment to the United States of not only HEU and LEU spent nuclear fuel, but of residual material from the production of molybdenum-99 for medical purposes. Molybdenum-99 is produced by the irradiation of targets in a research reactor. The targets are physically similar to the fuel for foreign research reactors. After being irradiated in a reactor, the targets are dissolved in acid to recover the molybdenum, leaving residual material containing enriched uranium. The United States has supplied HEU to Canada, Belgium, Argentina, and Indonesia for use as targets in the production of medical isotopes. The NRU reactor in Canada produces nearly all radioisotopes used in nuclear medicine in the United States.

This subalternative involves the acceptance of the following amounts of target material from these countries:

Canada	0.525 MTHM
Belgium	0.029 MTHM
Argentina	0.0011 MTHM
Indonesia	0.0014 MTHM
Total	0.5565 MTHM

This total has been rounded up to 0.6 MTHM for the purpose of analysis in this EIS. Under this subalternative, about 216 kg (476 lb) of HEU from target material would be removed from international commerce. This would be in addition to the estimated 4.6 metric tons (5.1 tons) of HEU that would be removed from international commerce under the basic implementation of Management Alternative 1.

Because the residual material contains weapons-usable HEU, there is a strong nuclear weapons nonproliferation rationale for including it in the scope of the management policy. This course of action would be desirable from a nuclear weapons nonproliferation standpoint, since it would leave the United States in control of the disposition of foreign research reactor spent nuclear fuel containing HEU, as well as residuals from the production of molybdenum-99, thereby minimizing the risk that such material might be diverted to a nuclear weapons program. This subalternative removes the most HEU from international civil commerce and provides the most support to U.S. nuclear weapons nonproliferation policy.

Furthermore, this subalternative would give the molybdenum-99 producers an incentive to switch from HEU targets to LEU targets. Appropriate LEU targets are currently under development as part of the RERTR program, and this target material would be accepted under this subalternative subject to the same conditions as the LEU foreign research reactor spent nuclear fuel in the basic implementation of Management Alternative 1.

The target material may be transported in one of two solid powder forms—as a calcine or an oxide. The calcine form would require about 140 cask shipments, while the oxide form would require about 57 cask shipments. The incident-free and accident risks are different for each form. The calcine material would produce an estimated 2.5 times more incident-free risk, but an estimated 10 times less accident risk than the oxide material. Furthermore, for transporting target material (unlike spent nuclear fuel), the accident risks would be greater than the incident-free risks. Therefore, to estimate conservative radiological risks, DOE and the Department of State assumed the target material would be transported as an oxide powder.

Marine Transport and Port Activities Impacts

The acceptance of target material would cause a very minor change in the marine and port incident-free impacts calculated for the basic implementation of Management Alternative 1. Up to only 7 cask shipments of oxide target material (6 from Belgium and 1[1] from Argentina or Indonesia), excluding the shipments from Canada, are estimated to be needed. This is less than one percent of the marine cask shipments of all foreign research reactor spent nuclear fuel in the basic implementation of Management Alternative 1. The incident-free impact per shipment is also reduced because the dose rate resulting from a cask loaded with the target material is expected to be much lower than that resulting from a cask loaded with foreign research reactor spent nuclear fuel.

For accident conditions, DOE and the Department of State estimated the risk due to an accident in an east coast port. The risk during marine transport would be much lower than the risk during port activities. The population risk due to accidents during port activities with seven casks of oxide target material is estimated to be 3.2×10^{-9} LCF. This is much lower than the population risk due to accidents with the foreign research reactor spent nuclear fuel.

The MEI risk is estimated to be 2.9×10^{-10} LCF, which is somewhat higher than the corresponding risk for the foreign research reactor spent nuclear fuel, but still very low.

1 *Argentina or Indonesia would not produce enough target material to fill a transportation cask. In all likelihood, the target material from these countries would be shipped along with research reactor spent nuclear fuel elements.*

Ground Transport Impacts

Impacts of Incident-Free Ground Transport

The impacts of incident-free ground transportation of target material were analyzed in the same manner as for the basic implementation of Management Alternative 1, except that, based on the low activity of the target material, the maximum dose rate at a distance of 2 m (6.6 ft) from the vehicle is estimated to be 0.1 mrem per hour. The risks calculated in this section could be added to those associated with foreign research reactor spent nuclear fuel transport. The incident-free transportation of target material was estimated to result in total latent fatalities that ranged from 0.0002 to 0.003 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew. When the risks of transporting target material are added to the risks of transporting the foreign research reactor spent nuclear fuel, the highest estimate of the population risk is 0.30 LCF.

The range of fatality estimates was due to two factors: the option of using truck or rail to transport target material and combinations of Phase 1 and Phase 2 sites that created varying shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.00007 to 0.00074. The estimated number of radiation-related LCF for the general population ranged from 0.00015 to 0.0023, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0001 to 0.004.

The impacts of transportation related to target material are summarized in Figures 4-10 through 4-13 and are described in more detail in Appendix E.

Impacts of Accidents During Ground Transport

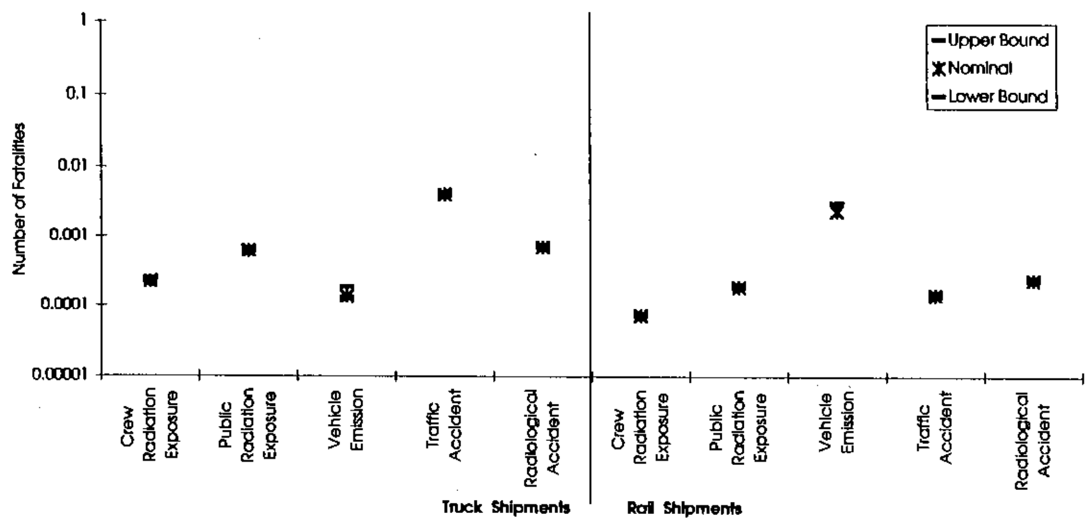
Cumulative transportation accident risks for the target material program are estimated to range from 0.0002 to 0.0054 LCF from radiation and from 0.0001 to 0.013 for traffic fatality, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation. The highest estimate of the population risk due to accidents involving target material (0.0054 LCF) is higher than the same risk involving foreign research reactor spent nuclear fuel (0.00028 LCF). This difference is due to the physical/chemical forms of the two substances. Adding these two risks together yields the population risk due to accidents under Implementation Subalternative 1c, 0.0057 LCF.

The maximum foreseeable offsite transportation accident involves a cask shipment of powderized target material in a suburban population zone, and the risk is estimated to be 9.3×10^{-11} LCF to the MEI.

The impacts of transportation accidents are summarized in Figures 4-10 through 4-13, as described in the previous section, and are described in more detail in Appendix E. These tables can be used to assess the bounded absolute and relative risk of this subalternative under each representative Programmatic SNF&INEL Final EIS alternative.

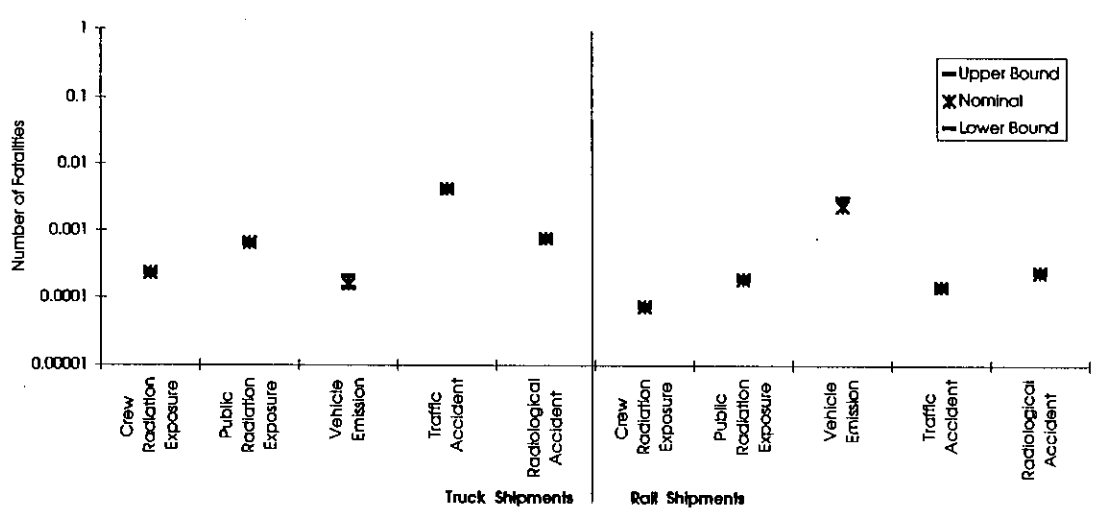
Management Site Impacts

There are two methods of preparing target material for transport. The first is calcining and canning the material with the aluminum included, and the second is to remove the aluminum from the solution, then oxidize and can the residue. Canned material from the first process has similar behavior as that of



- *Same values apply to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives*
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-10 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Decentralization Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-11 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative

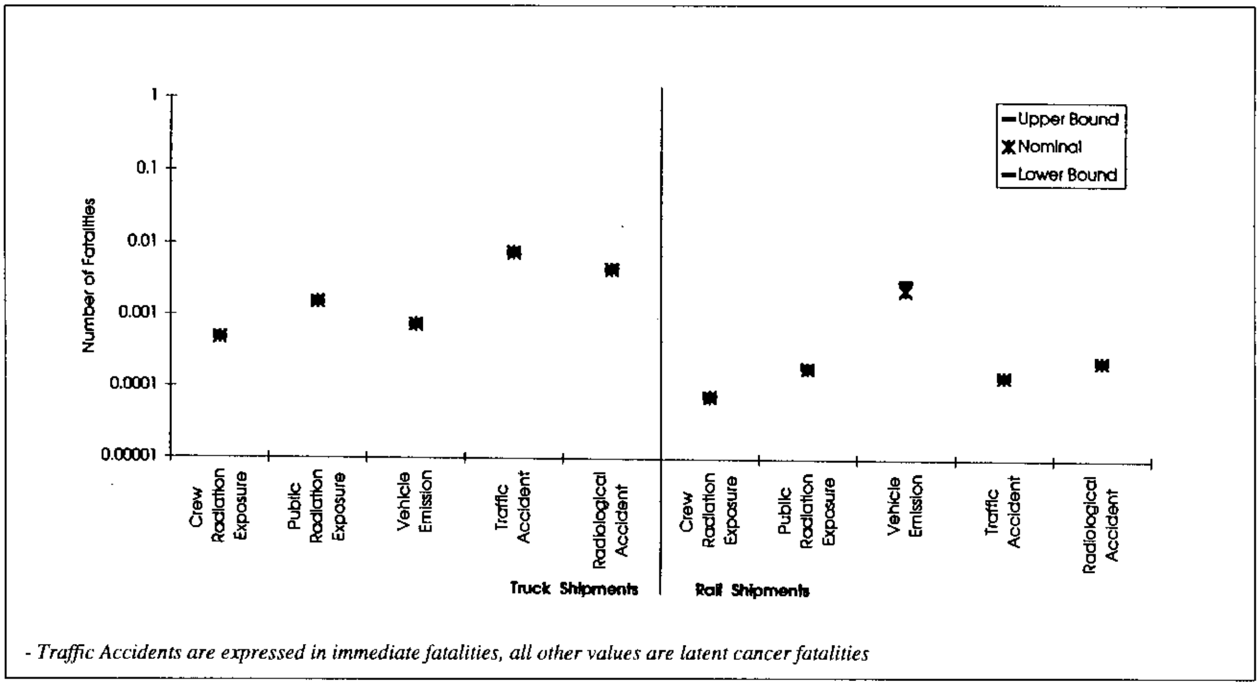


Figure 4-12 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative

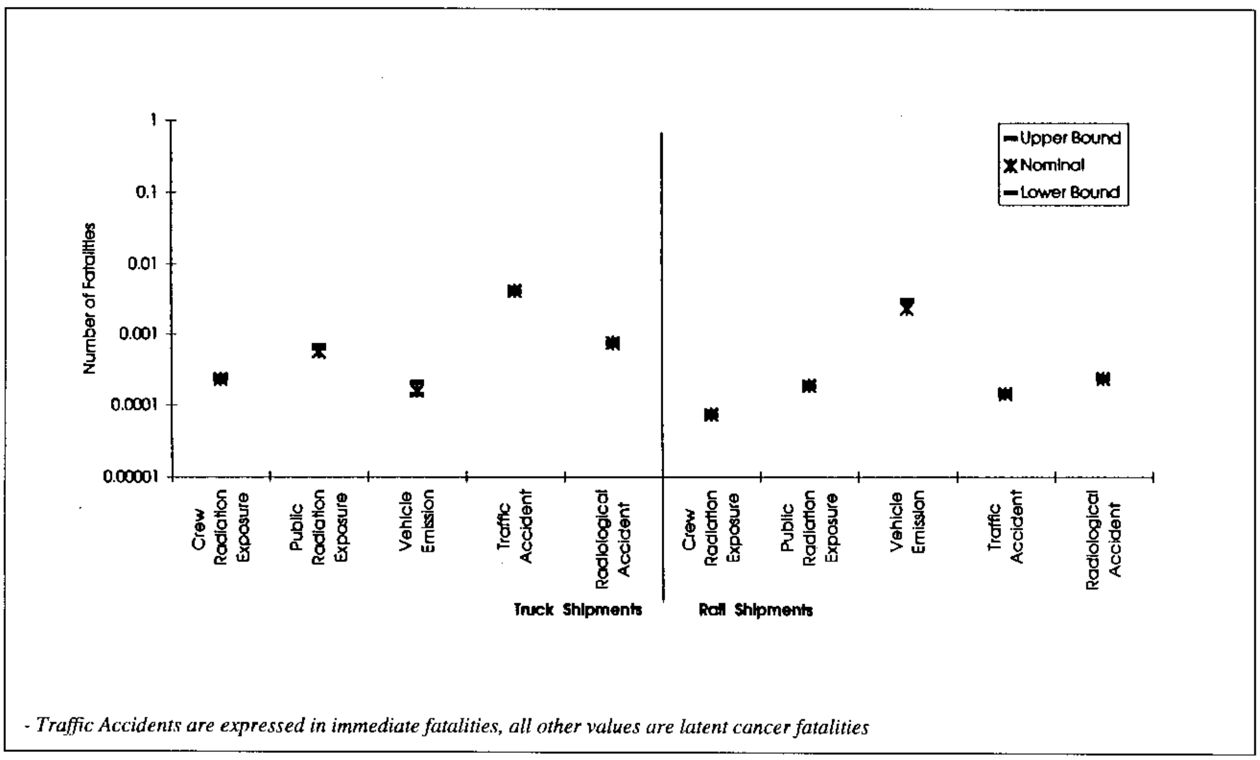


Figure 4-13 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

aluminum-based foreign research reactor spent nuclear fuel containing about 40 g of uranium per can. The second process allows a higher amount of uranium, about 200 g, to be packed in the same size can. Use of the first process would result in 6,750 cans representing approximately 140 cask shipments. The second process would result in 1,350 cans representing approximately 57 cask shipments.

Target material cans would be stored like foreign research reactor spent nuclear fuel elements. The storage space required is a function of volume rather than the nuclear or thermal characteristics of the target material. On average, four cans of target material could be stored in the same space as one foreign research reactor spent nuclear fuel element. Therefore, the maximum storage required for target material (in the 40-gram cans) would be equivalent to 1,700 foreign research reactor spent nuclear fuel elements or approximately 7.4 percent of the space required for the foreign research reactor spent nuclear fuel elements under the basic implementation of Management Alternative 1. The storage facilities analyzed for the basic implementation of Management Alternative 1 include this margin in the sizing.

Impacts of Incident-Free Management Site Activities

Radioactive emissions would not be expected from the target material receipt or storage because this material contains no gaseous fission products. Therefore, the incident-free radiological impacts to the public would be the same as in the basic implementation of Management Alternative 1.

The collective dose to the crews that would handle the cask shipments would be 70 person-rem, assuming that the cans from 140 cask shipments would be placed in dry storage casks. The associated worker population risk would be 0.03 LCF. Adding this risk to the worker population risk of the basic implementation of Management Alternative 1 yields 0.24 LCF for the total incident-free worker population risk for Implementation Subalternative 1c.

Impacts of Accidents Onsite

The process by which target material is prepared for shipment (i.e., drying and canning of the solutions, see Appendix B, Section B.1.5) releases all gaseous fission products. In addition, the cans do not require any trimming when they arrive at a storage facility. A review of the hypothetical accident scenarios in the basic implementation of Management Alternative 1 indicates that only the aircraft crash with fire accident scenario would be applicable to target material. The cans are never cut, and there are no gaseous fission products, so the foreign research reactor spent nuclear fuel elements breach scenario would not be applicable. In addition, should an aircraft crash into the wet storage pool where the target material is stored, or if an accidental criticality in the pool were to occur, the radioactivity releases would be bounded by those of the spent nuclear fuel analyzed for these accidents. This is because the radioactive inventory per can is very small compared to that in the bounding foreign research reactor spent nuclear fuel.

A scenario involving an aircraft crash into a dry storage facility with an ensuing fire was analyzed for the target material. The scenario assumptions are similar to those described in Appendix F, Section F.6. Because of the size of each can, it was assumed that the transfer cask involved in the accident would contain 40 cans of target material containing maximum radionuclide inventories, i.e., that of 40 cans of 200 g of uranium per can cooled for at least 3 years.

The frequency of this event is estimated to be 3 percent of the 1×10^{-6} per year used in the accident analysis of the basic implementation of Management Alternative 1. This is because the number of transfer casks involving target material is less than 3 percent of that used for the approximately 22,700 elements in the basic implementation of Management Alternative 1. Therefore, the frequency of this scenario is less than 10^{-7} per year, and is considered to be non-foreseeable. Nonetheless, this accident was analyzed and its frequency is set conservatively at 10^{-7} per year. The analytical procedure was the same as that used in the basic implementation of Management Alternative 1.

The highest estimate of the MEI/NPAI accident risk with target material is 2.0 x 10^{-10} LCF, which would occur at the Oak Ridge Reservation (Table F-118, Appendix F). This risk is lower than the highest MEI/NPAI risk in the basic implementation of Management Alternative 1 (0.000010 LCF), so the risk for this subalternative is the same as in the basic implementation of Management Alternative 1. This hypothetical individual would still have one chance in one hundred thousand of incurring an LCF due to an accident on a site.

The highest estimate of the population risk with target material is 1.9 x 10^{-7} LCF, which also would occur at the Oak Ridge Reservation (Table F-118, Appendix F). To obtain the total population risk for this subalternative, this risk must be added to the corresponding risk from the basic implementation of Management Alternative 1 (0.11 LCF). The population risk due to accidents with target material is so small compared to the risk due to the foreign research reactor spent nuclear fuel that it makes essentially no contribution to the population risk for this subalternative. The population risk due to accidents under this subalternative would be the same as that under the basic implementation of Management Alternative 1.

Summary of the Impacts of Implementation Subalternative 1c

The principal impacts under this subalternative would be occupational and public health and safety impacts. These are presented in Table 4-37 in terms of the risk of death due to cancer during each of the four segments of this subalternative. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. The impacts of the basic implementation of Management Alternative 1 (Table 4-35) are added to the impacts of managing the target material to obtain the impacts of this subalternative. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-37 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 1.4 x 10^{-7} LCF.

The highest estimated accident MEI risk is 0.000010 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

Table 4-37 Maximum Estimated Radiological Health Impacts of Implementation Subalternative 1c (Target Material)

	Risks (LCF)		
	Maximum Exposed Worker, MEI, or NPAI	Population	
		General Public	Workers
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 0.000029	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2.9×10^{-10}	0.000029	---
Ground Transport			
Incident-Free	0.00052	0.22	0.072
Accidents	9.3×10^{-11}	0.0057	---
Site Activities			
Incident-Free	0.026	0.00027	0.24
Accidents	0.000010	0.11	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000010	----	----
Total Population Risk			
Incident-Free	----	0.22	0.36
Accidents	----	0.12	----

As shown in Table 4-37, the total incident-free population risk would be 0.22 LCF for the potentially exposed public, while the corresponding risk would be 0.36 LCF for workers. Thus, there would be an estimated 22 percent chance of incurring one additional LCF among the exposed general public, and a 36 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-37. There is about a 15 percent chance that a truck driver or member of the public could die in a traffic accident associated with this subalternative. This death would be unrelated to the radioactive nature of the cargo.

4.3.2 Implementation Alternative 2: Alternative Policy Durations

DOE and the Department of State evaluated the impacts for two different policy durations under this implementation alternative: reducing the policy duration to 5 years and continuing the policy for HEU indefinitely.

4.3.2.1 Implementation Subalternative 2a: Five-Year Policy

Policy Considerations

Under this implementation subalternative, DOE would accept up to about 13 MTHM and about 18,800 elements of foreign research reactor spent nuclear fuel. This subalternative would reduce the number of foreign research reactor spent nuclear fuel elements that would be accepted by the United States to about 83 percent of the amount covered by the basic implementation of Management Alternative 1, and would accelerate the time at which the foreign research reactor operators and the governments of their host countries would become responsible for disposal of their own spent nuclear fuel. Up to about 4.1 metric tons (4.5 tons) of HEU would be removed from international commerce, which is about 0.5 metric tons (0.6 tons) less than under the basic implementation of Management Alternative 1.

This subalternative probably would not provide enough time for the foreign countries, especially the developing countries, to make arrangements for alternate means of managing their spent nuclear fuel. This could pressure various foreign research reactor operators to switch their reactors back to HEU fuel. In addition, it would probably, in effect, force many of the foreign research reactors with lifetime cores to shut down prematurely because it would be very difficult for them to find any means to dispose of their foreign research reactor spent nuclear fuel, other than to have DOE accept it.

Marine Transport Impacts

Impacts of Incident-Free Marine Transport

The impacts of incident-free marine transportation in the 5-year acceptance case were analyzed in the same manner as for the basic implementation of Management Alternative 1. The analysis was performed using the dose rates based on the exclusive-use regulatory limit for the shipment of spent nuclear fuel casks. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.025 to 0.028 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the ships' crews.

The range of impacts results from the analysis of shipment of the spent nuclear fuel on regularly scheduled commercial breakbulk vessels and on chartered container vessels, and would be the same as for vessels analyzed in the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the difference between the two estimates is a result of the shorter vessel journey time for chartered vessels due to the intermediate port stops associated with the regularly scheduled commercial transport of the spent nuclear fuel.

The maximally exposed worker risk would be lower than that in the basic implementation of Management Alternative 1 due to the reduced acceptance period. The highest estimated maximally exposed worker risk would be 0.00032 LCF.

Impacts of Accidents During Marine Transport

The consequences of the at-sea accidents for Implementation Subalternative 2a are no different than the consequences of at-sea accidents associated with the basic implementation of Management Alternative 1. For an accident involving the loss of a transportation cask in coastal waters the maximum exposure to an individual is estimated to be 14,000 mrem per year. DOE and the Department of State would mitigate this impact, however, by recovering the cask. Due to the reduced number of cask shipments, the likelihood of such an accident would be reduced. Under this subalternative, approximately 81 percent of the total number of cask shipments required under the basic implementation of Management Alternative 1 would be needed. The highest risk to a human, expressed in terms of peak dose rate, would be 0.00015 mrem per year from the loss of a damaged cask in the deep ocean. Assuming an individual receives this dose for 5 years, the total MEI risk would be about 4×10^{-10} LCF.

Port Activity Impacts

Impacts of Incident-Free Port Activities

In the analysis of the basic implementation of Management Alternative 1, the radiological impact of port activities was estimated on a per-shipment basis. Implementation Subalternative 2a results in approximately 81 percent of the total number of cask shipments that are required in the basic implementation of Management Alternative 1. The incident-free impacts of the port activities would be proportionally reduced. The estimated number of LCF associated with this subalternative ranges from 0.0027 to 0.0098. As in the marine incident-free analysis, this range of impacts is the result of the analysis of two modes of spent nuclear fuel shipment, regularly scheduled commercial breakbulk vessels and chartered container vessels.

The maximally exposed worker risk would be lower than that in the basic implementation of Management Alternative 1 due to the reduced acceptance period. The estimated maximally exposed worker risk would be 0.00032 LCF.

Impacts of Accidents During Port Activities

Port accident risks were calculated based on the per-shipment risks determined in the analysis of the basic implementation of Management Alternative 1. The analysis examined the impact of using a wide range of ports based on the population around the port city, from high density population ports such as Elizabeth, NJ, to low-density ports such as the MOTSU terminal in North Carolina. The analysis also considered the impact of chartered shipments (no intermediate port stops before the vessel reaches the spent nuclear fuel port of entry) versus regularly scheduled commercial shipments with up to two intermediate ports of call before the spent nuclear fuel port of entry. The port accident risks over the entire program are estimated to range from 3×10^{-7} to 0.00002 LCF from radiation. The range of fatality estimates is due to both the differences in port city populations and the number of intermediate port stops.

The MEI risk would be lower than that of the basic implementation of Management Alternative 1 due to the reduced number of cask shipments. The highest estimated MEI risk is 1.6×10^{-10} LCF.

Ground Transport Impacts

Impacts of Incident-Free Ground Transport

The impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.010 to 0.27 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates was due to two factors: the option of using truck or rail to transport spent nuclear fuel and combinations of Phase 1 and Phase 2 management sites that created varying shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.005 to 0.064. The estimated number of radiation-related LCF for the general population ranged from 0.005 to 0.20, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.001 to 0.041.

The maximally exposed worker risk would be lower than that in the basic implementation of Management Alternative 1 due to the reduced acceptance period. The highest estimated MEI risk would be 0.00032 LCF.

The impacts of transportation are summarized in Figures 4-14 through 4-17 and are described in more detail in Appendix E.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000003 to 0.00026 LCF from radiation and from 0.001 to 0.13 for traffic fatality, depending on the transportation mode and the potential foreigu research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as described for incident-free transportation.

The consequences of the maximum foreseeable offsite transportation accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced amount of ground transport, so the MEI risk is reduced to 1.1×10^{-11} LCF.

The impacts of transportation accidents are summarized in Figures 4-14 through 4-17, as described in the previous section, and are described in more detail in Appendix E. These tables can be used to assess the bounded absolute and relative risks of this subalternative under each representative Programmatic SNF&INEL Final EIS alternative.

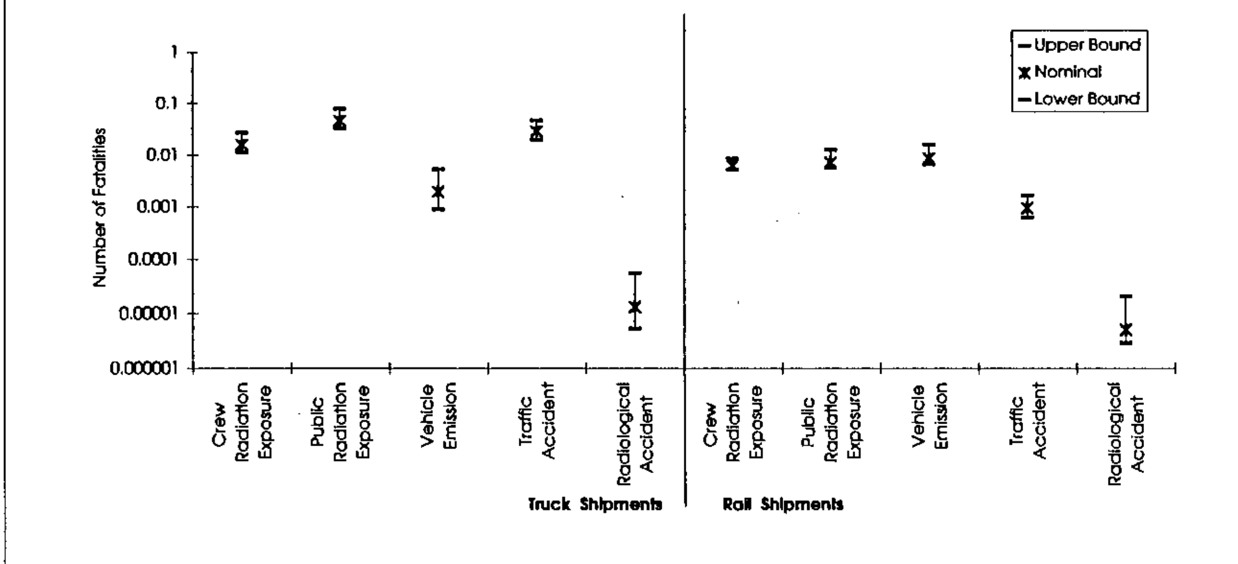
Management Site Impacts

As discussed in Chapter 2 of this EIS, Implementation Subalternative 2a reduces the quantity of foreign research reactor spent nuclear fuel to be managed to approximately 18,800 elements (compared to approximately 22,700 in the basic implementation of Management Alternative 1), but increases the rate of receipt to about 2,350 elements per year for an 8-year receipt period. This rate could challenge the capability of handling the incoming foreign research reactor spent nuclear fuel at a single site and could necessitate the use of both the Idaho National Engineering Laboratory and the Savannah River Site as near term foreign research reactor spent nuclear fuel management sites.

Incident-Free Impacts

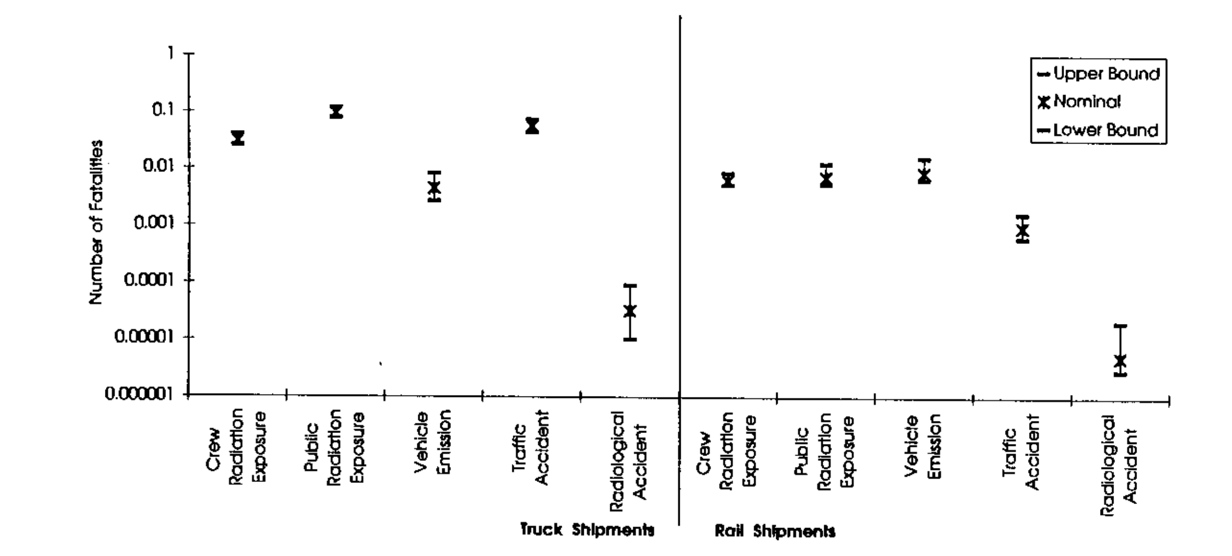
Based on the reduced number of foreign research reactor spent nuclear fuel elements that would be accepted under this subalternative, the worker population risk would be about 83 percent of that calculated for the basic implementation of Management Alternative 1. The maximally exposed worker risk was calculated in the same way as for the basic implementation of Management Alternative 1, with reduced handling time. If one worker received the maximum dose every year for eight years, his increased risk would be 0.016 LCF.

Some of the incident-free public risk depends on the amount of foreign research reactor spent nuclear fuel involved and some depends on the duration of each activity. The risk that accrues during receipt and handling can be scaled down by the factor of 83 percent from the basic implementation of Management Alternative 1, while the risk that accrues during storage is dependent only on the duration of the storage. The highest estimated incident-free public MEI risk in the basic implementation of Management Alternative 1 (1.4×10^{-7} LCF) is due to receipt and handling, so it is reduced by the factor of 83 percent to yield the corresponding risk for this subalternative (1.2×10^{-7} LCF).



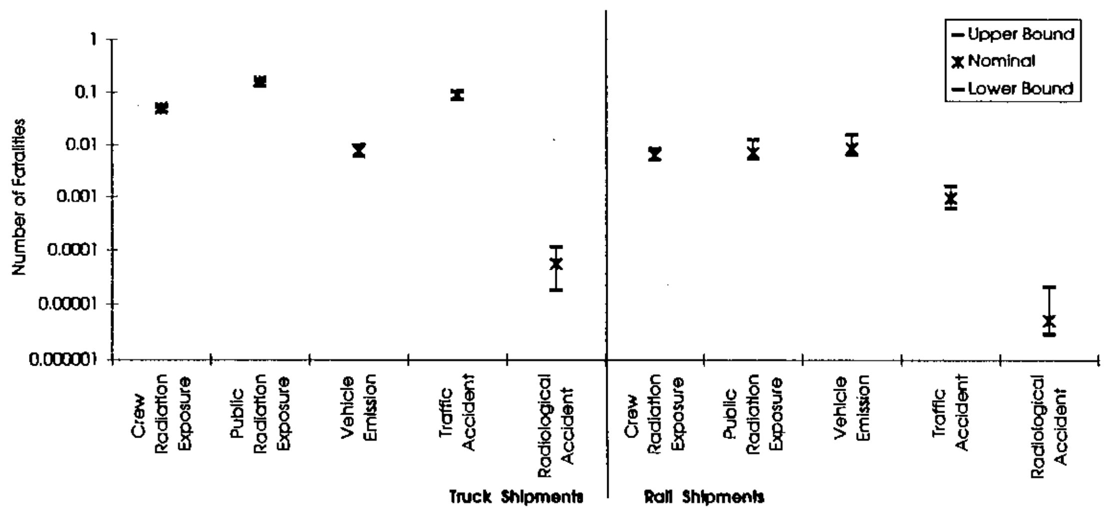
- *Same values apply to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives*
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-14 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Decentralization Alternative



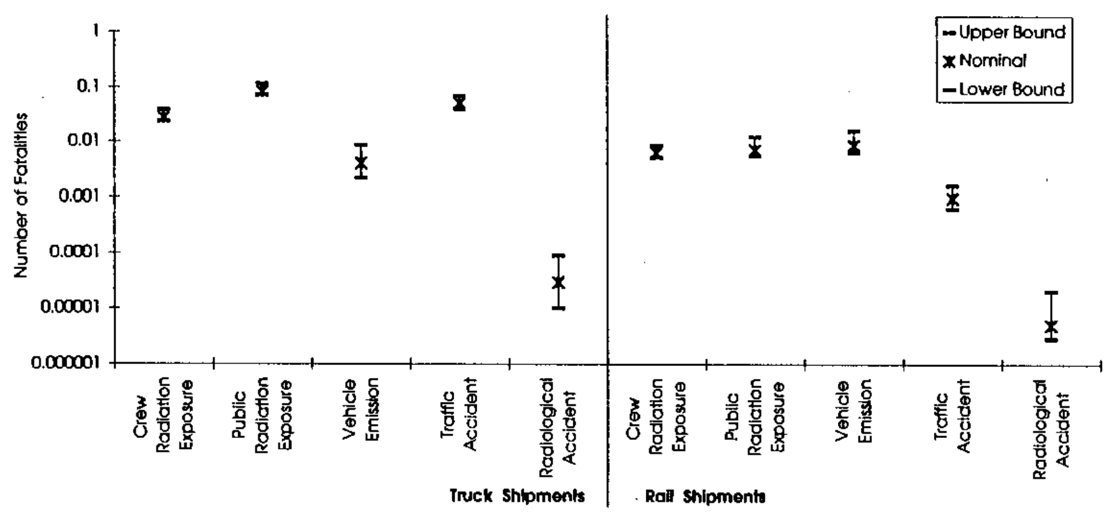
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-15 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-16 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-17 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

The highest estimated incident-free public population risk in Phase 1 of the basic implementation of Management Alternative 1 (0.00014 LCF) is due to 13 years of storage in L-Reactor Basin. The Phase 1 storage time in this subalternative would be slightly lower, and the estimated risk could be reduced, but for simplicity and to be conservative, DOE and the Department of State did not reduce this component of the risk estimate compared to the basic implementation. The corresponding Phase 2 risk (0.00013 LCF) is due to receipt and handling, so this component of the risk is reduced to 0.00011 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.00025 LCF.

Impacts of Accidents Onsite

The highest estimated public MEI risk due to accident conditions in the basic implementation of Management Alternative 1 (0.000010 LCF) is due to receipt and handling, so it is reduced by the factor of 83 percent to yield the corresponding risk for this subalternative (0.0000083 LCF). This is higher than any other combination of Phase 1 or Phase 2 annual risk and duration.

The highest estimated population risk due to Phase 1 accidents in the basic implementation of Management Alternative 1 (0.096 LCF) is due to an accidental criticality in RBOF. This facility would be used for less time in this subalternative and the estimated risk could be reduced, but for simplicity and to be conservative, DOE and the Department of State did not reduce this component of the risk estimate compared to the basic implementation. The corresponding Phase 2 risk (0.013 LCF) is due to receipt and handling, so this component of the risk is reduced by the factor of 83 percent, down to 0.011 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.11 LCF.

Summary of the Impacts of Implementation Subalternative 2a

The principal impacts under this subalternative would be occupational and public health and safety impacts. These are presented in Table 4-38 in terms of the risk of death due to cancer during each of the four segments of this subalternative. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-38 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

The highest estimated incident-free individual risk is 0.016 LCF, which would apply to an onsite radiation worker. This individual would have a 1.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 1.2×10^{-7} LCF.

The highest estimated accident MEI risk is 0.0000083 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in one hundred thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

Table 4-38 Maximum Estimated Radiological Health Impacts of Implementation Subalternative 2a (Five-Year Policy)

	Risks (LCF)		
	Maximum Exposed Worker, MEI, or NPAI	Population	
		General Public	Workers
Marine Transport			
Incident-Free	0.00032	0	0.028
Accidents	4×10^{-10}	much less than 0.00002	---
Port Activities			
Incident-Free	0.00032	0	0.0098
Accidents	1.6×10^{-10}	0.00002	---
Ground Transport			
Incident-Free	0.00032	0.20	0.064
Accidents	1.1×10^{-11}	0.00026	---
Site Activities			
Incident-Free	0.016	0.00025	0.17
Accidents	0.0000083	0.11	---
Highest Individual Risk			
Incident-Free	0.016	---	---
Accidents	0.0000083	---	---
Total Population Risk			
Incident-Free	---	0.20	0.27
Accidents	---	0.11	---

As shown in Table 4-38, the total incident-free population risk would be 0.20 LCF for the potentially exposed public, while the corresponding risk would be 0.27 LCF for workers. Thus, there would be an estimated 20 percent chance of incurring one additional LCF among the exposed general public, and a 27 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-38. There is about a 13 percent chance that a truck driver or member of the public could die in a traffic accident associated with this subalternative. This death would be unrelated to the radioactive nature of the cargo.

4.3.2.2 Implementation Subalternative 2b: Indefinite HEU/10-Year LEU Policy

Policy Considerations

The only difference between Implementation Subalternative 2b and the basic implementation of Management Alternative 1 would be to allow the acceptance of HEU spent nuclear fuel indefinitely from reactors with long-term lifetime cores, or from reactors whose operators for some reason (e.g., political) refuse to send their HEU spent nuclear fuel to the United States at this time. The exclusion of foreign research reactors that could be converted, but are not converted would be the same as in the basic implementation of Management Alternative 1. The amount of HEU spent nuclear fuel involved would also be the same as in the basic implementation of Management Alternative 1—only the timing would be different. The amount of HEU spent nuclear fuel that would be accepted after the policy period cannot be quantified because DOE and the Department of State do not know with certainty which countries would refuse to send their foreign research reactor spent nuclear fuel to the United States during the policy period of the basic implementation of Management Alternative 1. Nevertheless, this subalternative would provide a mechanism whereby DOE and the Department of State could increase the amount of U.S. origin HEU that could be recovered.

Impacts

The environmental impacts would be the same as, or slightly less than, those of the basic implementation of Management Alternative 1. Delaying the acceptance of a small fraction of the total amount of foreign research reactor spent nuclear fuel accepted would have a miniscule effect on the results presented in Section 4.2.

4.3.3 Implementation Alternative 3: Alternative Financing Arrangements

Under the basic implementation of Management Alternative 1, DOE and the Department of State would subsidize developing nations and charge developed nations a competitive rate. As discussed in Chapter 2, DOE and the Department of State have identified three potential financial arrangements:

- Subsidize all nations,
- Charge all nations the full cost of managing their spent nuclear fuel, and
- Subsidize developing nations and charge developed nations the full cost of managing their spent nuclear fuel.

Policy Considerations

Subsidizing all countries would be the most expensive for the United States. All the costs of transport, handling, storage, preparation for disposal, and disposal would be borne by the United States. The amount of HEU that would be accepted under this arrangement would likely be the same as under the basic implementation of Management Alternative 1.

Charging all countries the full cost of foreign research reactor spent nuclear fuel management would be the least expensive for the United States. All the costs would be borne by the foreign countries. Many developing countries probably would be unable to pay these high costs and this could lead to large quantities of HEU foreign research reactor spent nuclear fuel remaining in the countries least able to protect it. This could also lead to charges, rightly or wrongly, that the United States was not complying with its obligations under Article IV of the Non-Proliferation Treaty. Even some developed countries might refuse to pay a full cost recovery fee, thus broadening the scope of problems this arrangement could cause.

Subsidizing developing countries and charging developed countries full cost of spent nuclear fuel management would be somewhat less expensive for the United States than the basic implementation of Management Alternative 1. Developing countries would be treated the same as in the basic implementation of Management Alternative 1, but developed countries would be charged more than in the basic implementation of Management Alternative 1. It is not clear how much more because the amount of a full cost recovery fee cannot be determined accurately at this time. Nevertheless, this increase over the internationally competitive rate could lead those nations which can reprocess to do so and perhaps to switch back to HEU fuel. Those nations in which reprocessing is not a viable option might force their reactors to shut down, and then charge, rightly or wrongly, that the United States was not complying with its obligations under Article IV of the Non-Proliferation Treaty.

Impacts

The different financial arrangements under this implementation alternative would have no direct effect on the environmental impacts of accepting and managing foreign research reactor spent nuclear fuel. Indirect effects are possible because, if the price is too high, some reactor operators may choose not to ship their spent nuclear fuel to the United States. This would reduce the amount of spent nuclear fuel accepted and thereby reduce the environmental impacts. It would be speculative, at best, to estimate the amount of spent nuclear fuel that might be excluded under this implementation alternative compared to the basic implementation of Management Alternative 1, so the changes in the environmental impacts cannot be quantified. It is clear however, that these changes would reduce overall environmental impacts in the United States during the policy period.

4.3.4 Implementation Alternative 4: Alternative Locations for Taking Title

Policy Considerations

The Price-Anderson Act applies to the shipments, independent of who holds title to the spent nuclear fuel. Thus, there is no change in the liability protection provided to the citizens of the United States, no matter where DOE takes title to the foreign research reactor spent nuclear fuel. Hence, there would be no change in the physical mode of shipping nor in the cost of shipping. Nevertheless, DOE and the Department of State are considering the following arrangements regarding the location for taking title to the foreign research reactor spent nuclear fuel:

- Taking title prior to shipment [i.e., at the foreign research reactor(s)],
- Taking title at the port(s) of entry, and
- Taking title at the DOE management site(s).

If DOE were to take title to the foreign research reactor spent nuclear fuel at the foreign research reactors, the liability protection afforded the citizens of the United States would not change, and the shipping arrangements would still be the same. However, DOE would then be liable for any mishaps that might occur in the foreign nations, or on the high seas. Thus, the potential liability to the United States might exceed the liability under the basic implementation of Management Alternative 1.

Taking title at the port(s) of entry would leave title in the hands of the foreign research reactor operators for the distance from the U.S. territorial waters limit to the port, thus potentially causing public concern about who would be liable to respond to any accident that might occur during that portion of the shipment. Similarly, taking title at the DOE management site would leave title in the hands of the foreign research reactor operators for an even greater distance within the United States, leading to even greater public concerns. These potential concerns would be borne of a misunderstanding because ownership does not affect shipping arrangements and precautions or liability protection. Nevertheless, it is likely that such concerns would exist.

Impacts

The environmental impacts (if any) of spent nuclear fuel shipments are not affected by the identity of the owner of the spent nuclear fuel. Therefore, the point of transfer of title is not a factor in determining environmental impacts.

4.3.5 Implementation Alternative 5: Wet Storage Technology for New Construction

Wet storage technology for new construction was considered instead of the dry storage technology contained in the basic implementation of Management Alternative 1, for all five potential foreign research reactor spent nuclear fuel management sites. The impacts during marine transport, port activities, and ground transport would be the same as in the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the analysis examined environmental topics including land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, occupational health and safety, noise, utilities and energy, and waste management.

The means by which this alternative would be implemented at each management site are presented in Sections 2.6.5.3.1 through 2.6.5.3.5. The environmental impact analysis assumes that a new wet storage facility, which is described in Section 2.6.5.1.2, would be constructed at the sites to receive and store foreign research reactor spent nuclear fuel after the Phase 1 period. At the Savannah River Site, the alternative could also be implemented at the Barnwell Nuclear Fuels Plant (BNFP) and at the Hanford Site by the addition of facilities to the WNP-4 Spray Pond. These facilities are described in Appendix F, Section F.3. The analysis parallels in all respects the impact analysis performed for the new dry storage facility of the basic implementation of Management Alternative 1. It is presented in detail in Appendix F, Section F.4, with methodology and assumptions for radiological impacts given in Sections F.5 and F.6.

As in the basic implementation of Management Alternative 1, the analysis showed that this implementation alternative would not cause any major environmental impacts. Further, none of the environmental topics would clearly differentiate among the potential foreign research reactor spent nuclear fuel management sites.

4.3.5.1 Occupational and Public Health and Safety

As in the basic implementation of Management Alternative 1 (see Section 4.2.4.1) radiological exposures are presented as emissions-related impacts, handling-related impacts, and accident-related impacts.

Impacts to the Public of Incident-Free Management Site Activities

Table 4-39 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at each Phase 2 site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

The highest estimated Phase 1 public MEI and population risks for this alternative are identical to those for the basic implementation of Management Alternative 1. All possible Phase 1 MEI risks are lower than the highest estimated Phase 2 MEI risk in the next paragraph, so they will drop out. The highest Phase 1 component of the population risk is 0.00014 LCF in the basic implementation.

Among all the potential Phase 2 foreign research reactor spent nuclear fuel management sites, the maximum annual dose to the public from emissions is 0.06 mrem per year and 0.06 person-rem per year at the Oak Ridge Reservation for the MEI dose and the population dose, respectively. If it is assumed that receipt of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation could take place over a period of 3 years, the total MEI dose would be 0.18 mrem and the total population dose would be 0.18 person-rem. If it is further assumed that storage will continue for 30 years after the beginning of the receipt period, the total MEI dose from storage would be 1.4×10^{-5} mrem and the total population dose from storage would be 1.5×10^{-5} person-rem. The risks due to receipt and unloading would be much

Table 4-39 Annual Public Impacts for Receipt and Management of Foreign Research Reactor Spent Nuclear Fuel Under Implementation Alternative 5 (Wet Storage)

	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Savannah River Site				
Receipt/Unloading at:				
• BNFP	0.00065	3.3×10^{-10}	0.0045	0.0000023
• New Wet Storage Facility	0.00011	5.5×10^{-11}	0.0057	0.0000028
Storage at:				
• BNFP	7.5×10^{-9}	3.8×10^{-15}	4.8×10^{-8}	2.4×10^{-11}
• New Wet Storage Facility	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
Idaho National Engineering Laboratory				
Receipt/Unloading at:				
• New Wet Storage Facility	0.00038	1.9×10^{-10}	0.0031	0.0000016
Storage at:				
• New Wet Storage Facility	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
Hanford Site				
Receipt/Unloading at:				
• WNP-4 Spray Pond	0.00022	1.1×10^{-10}	0.0058	0.0000029
• New Wet Storage Facility	0.00020	1.0×10^{-10}	0.012	0.0000060
Storage at:				
• WNP-4 Spray Pond	5.9×10^{-10}	3.0×10^{-16}	1.6×10^{-8}	8.0×10^{-12}
• New Wet Storage Facility	8.8×10^{-10}	4.4×10^{-16}	6.9×10^{-8}	3.5×10^{-11}
Oak Ridge Reservation				
Receipt/Unloading at:				
• New Wet Storage Facility	0.060	3.0×10^{-8}	0.061	0.000031
Storage at:				
• New Wet Storage Facility	4.6×10^{-7}	2.3×10^{-13}	5.0×10^{-7}	2.5×10^{-10}
Nevada Test Site				
Receipt/Unloading at:				
• New Wet Storage Facility	0.00052	2.6×10^{-10}	0.00052	2.6×10^{-7}
Storage at:				
• New Wet Storage Facility	4.0×10^{-9}	2.0×10^{-15}	4.7×10^{-9}	2.4×10^{-12}

higher than those due to storage, so the maximum risk would be 0.18 mrem for the MEI and the sum of population doses would be 0.18 person-rem. The associated probabilities for incurring one LCF would be 9×10^{-8} LCF for the Phase 2 MEI risk and 0.00009 LCF for the Phase 2 population risk.

The maximum of the Phase 1 and Phase 2 incident-free public MEI risks is 9×10^{-8} LCF for this alternative. The sum of the Phase 1 and Phase 2 incident-free public population risks is 0.00023 LCF.

Impacts to Workers of Incident-Free Management Site Activities

As in the basic implementation of Management Alternative 1, workers would receive radiation doses during handling operations, such as receiving and unloading foreign research reactor spent nuclear fuel transportation casks at the site or transferring foreign research reactor spent nuclear fuel from one facility to another within the site. The methodology and assumptions for the analysis of this implementation alternative parallel that for the basic implementation of Management Alternative 1 as presented in Section 4.2.4.1 and Appendix F, Section F.5.

Table 4-40 presents the collective doses and risks that would be received by the members of the working crew, if that crew handled the total number of casks at the site.

Table 4-40 Handling-Related Impacts to Workers at Each Management Site Under Implementation Alternative 5 (Wet Storage)

Site	Worker Population Dose (person-rem)	Worker Population Risk (LCF)
Savannah River Site		
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phases 1 and 2: New Wet Storage Facility	360	0.14
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phases 1 and 2: BNFP	360	0.14
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phases 1 and 2: BNFP[a]	310	0.12
Idaho National Engineering Laboratory		
Phase 1: IFSF/CPP-749	257	0.10
Phases 1 and 2: New Wet Storage Facility	367	0.15
Phase 1: FAST	250	0.10
Phases 1 and 2: New Wet Storage Facility	360	0.14
Hanford Site		
Phase 2: New Wet Storage Facility or WNP-4 Spray Pond	109	0.04
Oak Ridge Reservation		
Phase 2: New Wet Storage Facility	109	0.04
Nevada Test Site		
Phase 2: New Wet Storage Facility	109	0.04

[a] *Assumes that BNFP would be ready in 5 years instead of 10 years.*

As seen from Table 4-40, the maximum total collective dose to workers handling foreign research reactor spent nuclear fuel at a single site would be 367 person-rem for the case analyzed at the Idaho National Engineering Laboratory, which assumes that all foreign research reactor spent nuclear fuel is in dry storage during Phase 1 and is transferred to a new wet storage facility for Phase 2. The associated probability for one LCF among the working crew would be 0.15. The highest dose to working crews for both phases in more than one site is 366 person-rem: 109 person-rem at one of the three Phase 2 sites plus 257 person-rem at the Idaho National Engineering Laboratory as the Phase 1 site. The associated probability for developing one LCF among the working crews of the two sites is 0.15.

Accident-Related Impacts

The accident scenarios analyzed for this implementation alternative are the same as those analyzed for the basic implementation of Management Alternative 1.

Table 4-41 presents the frequency and consequences of the accidents analyzed for each management site for this implementation alternative. Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at each candidate management site. Table 4-42 presents the annual risk estimates for wet storage.

The highest MEI or NPAI risk for Phase 1 would be the same as under the basic implementation of Management Alternative 1 (2.6×10^{-6} LCF). The highest annual MEI or NPAI risk for Phase 2 would be 0.000005 LCF per year, which is the annual risk to the NPAI from an accidental criticality at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be managed at the Oak Ridge Reservation for as long as 30 years, the Phase 2 component of this MEI/NPAI risk would

Table 4-41 Frequency and Consequences of Accidents at Each Management Site Under Implementation Alternative 5 (Wet Storage)

Site	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Savannah River Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.0070	3.5×10^{-9}	0.00039	2×10^{-10}	0.23	0.00012	0.14	5.6×10^{-8}
• Accidental Criticality	0.0031	17	0.0000085	9.5	0.0000048	370	0.19	1,600	0.00064
• Aircraft Crash	1×10^{-6}	4.1	0.0000021	0.98	4.9×10^{-7}	150	0.075	400	0.00016
BNFP									
• Spent Nuclear Fuel Assembly Breach[a]	0.16	0.018	9×10^{-9}	0.00099	5×10^{-10}	0.028	0.000014	0.00080	3.2×10^{-10}
• Accidental Criticality[a]	0.0031	80	0.000040	75	0.000038	44	0.022	75	0.000030
• Aircraft Crash	1×10^{-6}	92	0.000046	31	0.000016	23	0.012	70	0.000028
Idaho National Engineering Laboratory									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.0016	8×10^{-10}	0.0036	1.8×10^{-9}	0.43	0.00022	0.14	5.6×10^{-8}
• Accidental Criticality	0.0031	28	0.000014	30	0.000015	140	0.070	1,800	0.00072
• Aircraft Crash	1×10^{-6}	22	0.000011	9.8	0.0000049	250	0.13	400	0.00016
Hanford Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.13	6.5×10^{-8}	0.0033	1.7×10^{-9}	1.6	0.00080	0.25	1.0×10^{-7}
• Accidental Criticality	0.0031	64	0.000032	14	0.000007	740	0.37	3,600	0.0014
• Aircraft Crash[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
WNP-4 Spray Pond									
• Spent Nuclear Fuel Assembly Breach[a]	0.16	0.15	7.5×10^{-8}	0.0033	1.7×10^{-9}	1.3	0.00065	0.00024	9.6×10^{-11}
• Accidental Criticality[a]	0.0031	97	0.000049	76	0.000038	620	0.31	120	0.000048
• Aircraft Crash[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
Oak Ridge Reservation									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.71	3.6×10^{-7}	0.20	1.0×10^{-7}	16	0.0080	0.68	2.7×10^{-7}
• Accidental Criticality	0.0031	1,500	0.00075	3,300	0.0017	1,400	0.70	6,800	0.0027
• Aircraft Crash	1×10^{-6}	380	0.00019	600	0.00030	2,900	1.5	1,900	0.00076
Nevada Test Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.054	2.7×10^{-8}	0.0016	8×10^{-10}	0.33	0.00017	0.10	4.0×10^{-8}
• Accidental Criticality	0.0031	88	0.000044	15	0.0000075	54	0.027	1,300	0.00052
• Aircraft Crash	1×10^{-6}	29	0.000015	4.2	0.0000021	61	0.031	290	0.00012

[a] *Emissions would be released through a tall stack, so workers would receive low doses.*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

NA = Not applicable

Table 4-42 Annual Risks of Accidents at Each Management Site Under Implementation Alternative 5 (Wet Storage)

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Savannah River Site				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
• Accidental Criticality	2.7×10^{-7}	1.5×10^{-8}	0.00060	0.0000020
• Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
BNFP				
• Spent Nuclear Fuel Assembly Breach[a]	2.8×10^{-9}	8.0×10^{-11}	0.0000023	5.2×10^{-11}
• Accidental Criticality[a]	1.3×10^{-7}	1.2×10^{-7}	0.000070	9.2×10^{-8}
• Aircraft Crash	4.6×10^{-10}	1.6×10^{-11}	1.2×10^{-8}	2.8×10^{-10}
Idaho National Engineering Laboratory				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
• Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
• Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}
Hanford Site				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	1.1×10^{-8}	2.7×10^{-10}	0.00013	1.6×10^{-8}
• Accidental Criticality	1.0×10^{-7}	2.2×10^{-8}	0.0012	0.0000044
• Aircraft Crash[b]	NA	NA	NA	NA
WNP-4 Spray Pond				
• Spent Nuclear Fuel Assembly Breach[a]	1.2×10^{-8}	2.7×10^{-10}	0.00011	1.5×10^{-11}
• Accidental Criticality[a]	1.5×10^{-7}	1.2×10^{-7}	0.00096	1.5×10^{-7}
• Aircraft Crash[b]	NA	NA	NA	NA
Oak Ridge Reservation				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-8}	1.6×10^{-8}	0.0013	4.4×10^{-8}
• Accidental Criticality	0.0000024	0.000005	0.0022	0.0000084
• Aircraft Crash	1.9×10^{-10}	3.0×10^{-10}	0.0000015	7.6×10^{-10}
Nevada Test Site				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	4.2×10^{-9}	1.3×10^{-10}	0.000026	6.4×10^{-9}
• Accidental Criticality	1.4×10^{-7}	2.3×10^{-8}	0.000084	0.000016
• Aircraft Crash	1.5×10^{-11}	2.1×10^{-12}	3.1×10^{-8}	1.2×10^{-10}

[a] *Emissions would be released through a tall stack, so workers would receive low doses.*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

NA = Not applicable

be 0.00015 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations in this implementation alternative. Taking the maximum of the Phase 1 and Phase 2 MEI risks yields 0.00015 LCF for the maximum MEI risk due to accidents.

The highest population risk for Phase 1 would be the same as under the basic implementation of Management Alternative 1, 0.096 LCF. The highest annual population risk for Phase 2 would be 0.0022 LCF per year, which is the annual risk to the public from an accidental criticality at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be managed at the

Oak Ridge Reservation for as long as 30 years, the Phase 2 component of this population risk would be 0.066 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations in this implementation alternative. Adding the Phase 1 and Phase 2 population risks yields 0.16 LCF for the total population risk due to accidents.

4.3.5.2 Topics Not Discussed in Detail

Nonradiological impacts associated with the wet storage implementation alternative are similar to those for dry storage considered in the basic implementation of Management Alternative 1. They are discussed in detail in Appendix F, Section F.4.

Impacts at each management site typically associated with construction activities such as land use, socioeconomics, cultural resources, aesthetic and scenic resources, air quality, ecology, and noise are similar because: (1) both dry and wet storage facilities could be constructed at the same locations at each site; and (2) both facilities are approximately the same size. As indicated in Section 2.6.5.1, the construction of the wet storage facility would disturb approximately 2.8 ha (7 acres) of land while the construction of the dry storage facility would disturb 3.6 to 4.5 ha (9 to 11 acres). Specifically for the Savannah River Site, if the wet storage alternative is implemented using the BNFP facility there would be no impacts associated with construction activities.

Impacts at each management site typically associated with the operation of the facilities such as air quality, water quality, socioeconomics, utilities, and waste generation are also very similar as indicated in Section 2.6.5.1. The only notable difference is indicated in water use. The wet storage facility would use 1.5 million liters (409,000 gal) per year during the storage mode of the operation (over 30 years) compared to 0.9 million liters (238,000 gal) per year used by the dry storage facility over the same period. This difference, however, is small compared to typical water consumption rates at the sites: 1.14 billion liters (300 million gal) per year at the Nevada Test Site to 88 billion liters (23.2 billion gal) per year at the Savannah River Site.

4.3.5.3 Summary of the Impacts of Implementation Alternative 5

The principal impacts under this implementation alternative would be occupational and public health and safety impacts. These are presented in Table 4-43 in terms of the risk of death due to cancer during each of the four segments of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-43 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

Table 4-43 Maximum Estimated Radiological Health Impacts of Implementation Alternative 5 (Wet Storage)

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 0.000029	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2×10^{-10}	0.000029	---
Ground Transport			
Incident-Free	0.00052	0.22	0.071
Accidents	1.4×10^{-11}	0.00028	---
Site Activities			
Incident-Free	0.026	0.00023	0.15
Accidents	0.00015	0.16	---
Highest Individual Risk			
Incident-Free	0.026	---	---
Accidents	0.00015	---	---
Total Population Risk			
Incident-Free	---	0.22	0.27
Accidents	---	0.16	---

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have approximately a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 9×10^{-8} LCF.

The highest estimated accident MEI risk is 0.00015 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than two in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-43, the total incident-free population risk would be 0.22 LCF for the potentially exposed public, while the corresponding risk would be 0.27 LCF for workers. Thus, there would be an estimated 22 percent chance of incurring one additional LCF among the exposed general public, and a 27 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-43. There is about a 14 percent chance that a truck driver or member of the public could die in a traffic accident associated with this implementation alternative. This death would be unrelated to the radioactive nature of the cargo.

4.3.6 Implementation Alternative 6: Near Term Chemical Separation in the United States

As discussed in Section 2.2.2.6, this implementation alternative involves conventional chemical separation in existing facilities at either the Savannah River Site or the Idaho National Engineering Laboratory. The facilities at the Savannah River Site are limited to chemically separating the aluminum-based foreign research reactor spent nuclear fuel. After some upgrading, the facilities at the Idaho National Engineering Laboratory would have the capability to chemically separate all the foreign research reactor spent nuclear fuel.

4.3.6.1 Implications of Chemical Separation for U.S. Nonproliferation Policy

As a matter of policy, the United States does not currently engage in reprocessing or chemical separation to extract plutonium for civilian or military purposes. U.S. policy is also not to encourage the civilian use of plutonium and to explore means to limit the stockpiling of plutonium from civil nuclear programs. This alternative nonetheless considers scenarios whereby the United States might engage in future chemical separation of foreign research reactor spent nuclear fuel. If a decision were made pursuant to this EIS to chemically separate some or all of the foreign research reactor spent nuclear fuel, the limited amount of plutonium in the spent fuel would not be separated. Rather it would be left in, and disposed of with, the high-level radioactive wastes produced during the chemical separation operation.

Two alternatives are evaluated for handling the highly enriched uranium in the spent fuel, either to blend it down to low enriched uranium (the preferred alternative, if any chemical separation is undertaken), or to separate it as HEU and place it in safe, secure storage. Chemical separation of foreign research reactor spent nuclear fuel, with blending down of the separated uranium, would, in fact, result in a reduction in the amount of HEU – a major goal of the U.S. Nuclear Weapons Nonproliferation Policy announced in September 1993. Despite this fact, there is a concern that other states may perceive only that the U.S. has restarted reprocessing.

For example, the potential exists that other states (e.g., Iran), might use the restart of reprocessing in the United States as an excuse to continue current programs or begin new ones – activities that would run counter to U.S. nuclear weapons nonproliferation interests. The implications in North Korea, where the United States has been actively working to create a nonreprocessing zone, as well as in other states, could complicate current U.S. nonproliferation activities.

4.3.6.2 General Assumptions and Analytic Approach

Potential impacts at the Savannah River Site and the Idaho National Engineering Laboratory were estimated separately. The impacts due to chemical separation and associated onsite activities would be in addition to those due to marine transport, port activities, and ground transport.

As discussed in Section 2.2.2.6, DOE and the Department of State have analyzed four possible chemical separation subalternatives under this implementation alternative. These four subalternatives, with spent nuclear fuel amounts and estimated facility run durations are:

	Amount (MTHM)	***Duration (Years)***
Savannah River Site (only aluminum-based spent nuclear fuel)		
• Foreign research reactor spent nuclear fuel only	18.2	13
• Foreign research reactor spent nuclear fuel plus other spent nuclear fuel	51	13
Idaho National Engineering Laboratory (aluminum-based and TRIGA spent nuclear fuel)		
• Foreign research reactor spent nuclear fuel only	19.2	12
• Foreign research reactor spent nuclear fuel plus other spent nuclear fuel	65	12

The duration of chemical separation operations dedicated to foreign research reactor spent nuclear fuel is driven by the rate of foreign research reactor spent nuclear fuel receipt at the Savannah River Site or the Idaho National Engineering Laboratory. The facility run durations at Savannah River Site are both up to 13 years, whether the facilities would be chemically separating only the 18.2 MTHM of foreign research reactor spent nuclear fuel or the 51 MTHM of spent nuclear fuel. Because the additional spent nuclear fuel would be chemically separated at the same time as the foreign research reactor spent nuclear fuel in a parallel process, only the combined impacts will be used to determine the risks associated with the overall operations. There are other nuclear materials, such as the Mark-31 targets currently stored at the Savannah River Site, which could also be chemically separated. These nuclear materials are not included in this implementation alternative, but they are covered under cumulative impacts. The impacts of running the facilities are based on conservative assumptions regarding incident-free annual emissions and possible accident releases which cover this range of throughputs.

The facility run durations at the Idaho National Engineering Laboratory are estimated to be up to 12 years. Furthermore, the same type of conservative assumptions regarding incident-free emissions and accidental releases are applied to calculate the environmental impacts.

As discussed in Section 2.2.2.6, the implementation component of uranium disposition has policy implications. The separated LEU could be returned to the commercial sector for reuse as reactor fuel. The HEU could be blended down to LEU or it could be processed directly to an oxide and stored. If a decision is made to chemically separate this spent nuclear fuel, it would be DOE's preference to blend down the HEU to LEU and thus preclude the possibility of this material ever being used in a nuclear weapon.

4.3.6.3 Marine Transport Impacts

The marine transport impacts of this implementation alternative would be identical to those of the basic implementation of Management Alternative 1, as discussed in Section 4.2.1.

4.3.6.4 Port Activities Impacts

The port activities impacts of this implementation alternative would be identical to those of the basic implementation of Management Alternative 1, as discussed in Section 4.2.2.

4.3.6.5 Ground Transport Impacts

The impacts due to ground transport of foreign research reactor spent nuclear fuel in this implementation alternative would be slightly lower than those of the basic implementation of Management Alternative 1, because the Phase 2 intersite shipments would not occur.

If the aluminum-based foreign research reactor spent nuclear fuel were chemically separated at the Savannah River Site it could not then be transported to another management site as spent nuclear fuel. The high-level waste resulting from this chemical separation would be managed onsite for the duration of the 40-year program period. The TRIGA foreign research reactor spent nuclear fuel would be transported to either of the two sites for management and it would not be transported again for the duration of the 40-year program period.

Similarly, if all the foreign research reactor spent nuclear fuel were chemically separated at the Idaho National Engineering Laboratory, it could not then be transported to another management site as spent nuclear fuel. The high-level waste resulting from this chemical separation would be managed onsite for the duration of the 40-year program period.

Impacts of Incident-Free Ground Transport

The impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.020 to 0.27 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates was due to two factors: the option of using truck or rail to transport spent nuclear fuel and combinations of management sites that created varying cask shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.009 to 0.065. The estimated number of radiation-related LCF for the general population ranged from 0.011 to 0.21, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.003 to 0.05.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000004 to 0.00014 LCF from radiation and from 0.002 to 0.13 for traffic fatality, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation.

The consequences of the maximum foreseeable offsite transportation accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced amount of ground transport, so the MEI risk is reduced to 1.3×10^{-11} LCF.

4.3.6.6 Impacts at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

DOE and the Department of State evaluated near term chemical separation at the Savannah River Site and the Idaho National Engineering Laboratory for five key types of impacts: (1) Socioeconomics, (2) Air Quality, (3) Water Quality, (4) Occupational and Public Health and Safety, and (5) Waste Management. The other impacts are all the same as those described in the basic implementation of Management Alternative 1. The analytic approach was to use the results published in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and the Interim Management of Nuclear Materials Final EIS (DOE, 1995a) whenever possible. Usually, these results can be adopted directly.

4.3.6.6.1 Socioeconomics

Savannah River Site

The chemical separation facilities at the Savannah River Site were last operated in 1992. The facilities are in a warm standby condition and are currently fully staffed. Use of these facilities would not have a notable net impact upon employment or the regional economy.

Idaho National Engineering Laboratory

The chemical separation facilities at the Idaho National Engineering Laboratory were last operated in 1990 and are currently in the process of being cleaned out in preparation for decommissioning. Some staff would need to be added eventually, but the use of these facilities would not have a notable net impact upon employment or the regional economy.

4.3.6.6.2 Air Quality

Savannah River Site

Incident-Free Nonradiological Emissions

DOE has analyzed the expected nonradiological emissions from its chemical separations facilities at the Savannah River Site in the Programmatic SNF&INEL Final EIS (DOE, 1995c). All estimated emissions would be small increases over baseline site-wide totals and within regulatory limits (DOE, 1995c).

Incident-Free Radiological Emissions

DOE has analyzed the expected airborne radiological emissions from the Savannah River Site chemical separations facilities in the Interim Management of Nuclear Materials EIS (DOE, 1995a). These radiological emissions are presented in Table 4-44 (Grainger, 1995). The health effects from these airborne emissions are discussed in Section 4.3.6.6.4 below.

Table 4-44 Annual Incident-Free Airborne Radiological Emissions at the Savannah River Site that Contribute to the Offsite Dose[a]

Element	*Ci/yr*
Tritium	57.8
Cesium-134	0.002
Cesium-137	0.12
Curium-242/244	0.12
Cerium-144	0.0059
Americium-241	0.016
Cobalt-60	0.000000053
Plutonium-238	0.078
Plutonium-239	0.020
Strontium-89/90	0.17
Iodine-131	0.0053
Uranium-235/238	0.039
Antimony-125	0.018
Ruthenium-106	0.20

[a] *Krypton-85 would be released at an estimated rate of 120,000 Ci/yr*

Source: Grainger, 1995

Krypton-85 emissions are not included in Table 4-44 because these releases are not normally measured or calculated. The health effects resulting from krypton-85 releases are very low compared to those resulting from other isotopes that are being measured. Krypton is an inert gas with no affinity for biological systems, so it does not adhere to the lungs if inhaled. The radioactive isotope of krypton would cause such a low level of harm to the population near the Savannah River Site because it remains in the human body for only very brief periods of time. The total amount of krypton-85 that would be contained in all of the aluminum-based foreign research reactor spent nuclear fuel is conservatively estimated to be 1.5 x 10^6 curies. Assuming this is released gradually over the 12-year reprocessing period, the annual emission rate would be 1.2 x 10^5 curies per year.

Idaho National Engineering Laboratory

Incident-Free Nonradiological Emissions

DOE has analyzed the expected nonradiological emissions from its chemical separations facilities at the Idaho National Engineering Laboratory in the Programmatic SNF&INEL Final EIS (DOE, 1995c). All estimated emissions are within regulatory limits (DOE, 1995c).

Incident-Free Radiological Emissions

DOE has also analyzed the expected radiological emissions from the Idaho National Engineering Laboratory chemical separations facilities in the Programmatic SNF&INEL Final EIS (DOE, 1995c). These are presented in Table 4-45. The radiological emission rates were estimated using conservative engineering calculations based on knowledge of the proposed activity. These emission rates are representative of emissions that could occur during Implementation Alternative 6 at the Idaho National Engineering Laboratory. Human health consequences are discussed in Section 4.3.6.6.4.

Table 4-45 Annual Incident-Free Airborne Radiological Emissions at the Idaho National Engineering Laboratory

Element	Ci/yr
Tritium + Carbon-14	3,100
Cesium-134 + Cesium-137	0.18
Cobalt-60	0.0000019
Plutonium	0.0077
Strontium-90 + Yttrium-90	0.058
Krypton-85	500,000
Antimony-125	16
Iodine-129 + Iodine-131	0.44
Others	0.21

Source: DOE, 1995b

4.3.6.6.3 Water Quality

Savannah River Site

DOE has analyzed the expected liquid radiological releases from the Savannah River Site chemical separations facilities in the Interim Management of Nuclear Materials EIS (DOE, 1995a). These releases are presented in Table 4-46 (Grainger, 1995). The health effects from these liquid releases are discussed in Section 4.3.6.6.4 below.

Idaho National Engineering Laboratory

Chemical separation activities at the Idaho National Engineering Laboratory would not affect water quality because the facility designs would prevent any accidental or incident-free discharge of liquid effluents (DOE, 1995c).

Table 4-46 Annual Incident-Free Liquid Radiological Releases at the Savannah River Site

Element	*Ci/yr*
Tritium	1.29
Strontium-89/90	0.013
Ruthenium-103/106	0.012
Cesium-137	0.033
Promethium-147	0.045

4.3.6.6.4 Occupational and Public Health and Safety

Potential exposures to workers and the public due to chemical separation activities were analyzed at both the Savannah River Site and the Idaho National Engineering Laboratory (DOE, 1995c). To estimate health effects, this analysis defined three receptor groups:

- onsite workers assigned to operations involving spent nuclear fuel,
- 1994 offsite population residing within an 80-km (50-mi) radius of the chemical separation facilities (exposure via air), and
- offsite population whom management site surface-water emissions could affect.

Each of these three receptor groups would receive an annual maximum individual dose and an annual population dose. The maximally exposed worker dose would be limited by regulation to 5,000 mrem per year, as in the basic implementation of Management Alternative 1.

Savannah River Site

Incident-Free Impacts at the Savannah River Site

The highest estimated incident-free dose rates for conventional chemical separation operations at the Savannah River Site are presented in Table 4-47 (DOE, 1995a). These chemical separation operations could include activities related to blending the separated HEU down to LEU and converting all LEU into an oxide suitable for long-term storage. Values in Table 4-47 represent the estimated dose rates due to these activities, including actual chemical separation, blending down, and conversion to oxide. Multiplying these values by the estimated program duration of 13 years yields the doses presented in Table 4-48. These doses are converted into risks of LCF by applying the appropriate conversion factors and these results are also presented in Table 4-48. If the HEU were not blended down, but rather converted directly to oxide, the worker population dose would be higher because the conversion to oxide would take place in the Uranium Solidification Facility. In this facility, the workers would be closer to the uranium.

Table 4-47 Incident-Free Radiation Dose Rates Due to Chemical Separation at the Savannah River Site

	Maximum Individual Dose Rate (mrem/yr)	*Population Dose Rate (person-rem/yr)*
Public		
Via Air	0.66	27
Via Water	0.0098	0.033
Workers	5,000[a]	21

[a] *Assumed to be equal to the regulatory limit*

Table 4-48 Radiological Health Impacts Due to Incident-Free Chemical Separation Operations at the Savannah River Site

	Maximum Individual Dose (mrem)	*Maximum Individual Risk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Public				
Via Air	8.6	0.0000043	351	0.18
Via Water	0.13	6.4×10^{-8}	0.43	0.00021
Workers	65,000	0.026	273	0.11

These risks must be combined with the risks of receiving/unloading the casks. Risks to the public were presented earlier in this chapter in Table 4-8. The risks of storage at RBOF are also presented, but they are very low compared to those of receipt/unloading. Assuming the foreign research reactor spent nuclear fuel would be received at RBOF for the full 13 years, the public MEI and population risks would be 7.1×10^{-10} LCF and 0.000036 LCF, respectively. These risks are much lower than the corresponding values in Table 4-48.

The handling-related risks to workers were presented earlier in this chapter. Under the conservative assumptions in the basic implementation of Management Alternative 1, the maximally exposed worker risk due to handling could be as high as 0.026 LCF which is equal to the 0.026 LCF in Table 4-48.

For the public, the estimated MEI risk from incident-free chemical separation activities would be 0.0000043 LCF. This risk means that an individual who lives at the Savannah River Site boundary would have an additional chance of less than one in one hundred thousand of incurring an LCF.

The handling-related worker population risk at RBOF is 0.10 LCF (see Table 4-14). This must be added to the 0.11 LCF from Table 4-48 to obtain the estimate of worker population risk due to chemical separation of foreign research reactor spent nuclear fuel. The estimated population risk for workers, including the handling-related risk, is 0.21 LCF, so there would be an approximately 21 percent chance of one additional LCF among the radiation workers.

The estimated total public population risk from chemical separation activities would be 0.18 LCF (see Table 4-48), which means that there would be an approximately 18 percent chance of one additional LCF among the population residing around the Savannah River Site due to incident-free chemical separation activities.

Impacts of Chemical Separations Accidents at the Savannah River Site

DOE has analyzed the impacts of reasonably identifiable accidents due to chemical separation activities at the Savannah River Site (DOE, 1995a), including a hydrogen explosion in a high-level waste tank, an unpropagated fire in a solution vessel, two kinds of inadvertent transfers of solutions, a coil and tube failure in the cooling system, a nuclear criticality, a "red oil" explosion, a severe earthquake, and a tornado. The annual risks for the accident with the highest estimated combination of frequency and consequence are presented in Table 4-49. The most severe accident scenario is an unpropagated fire in a solution vessel. Multiplying these results by the estimated program duration of 13 years yields the risks presented in Table 4-50.

Table 4-49 Annual Impacts of Chemical Separation Accidents at the Savannah River Site

	Accident Frequency (per year)	Consequences (LCF)		Risks (LCF/yr)	
		Maximum Individual	Population	Maximum Individual	Population
Unpropagated Fire					
• Public	0.02	0.00018	1.3	0.0000036	0.026
• Workers	0.02	0.00086	---	0.000017	---

Table 4-50 Impacts of Accidents During Chemical Separation Operations at the Savannah River Site

	Maximum Individual Risk (LCF)	Population Risk (LCF)
Public	0.000047	0.34
Workers	0.00022	---

These results indicate that the estimated public MEI risk due to the chemical separation accidents is 0.000047 LCF. The estimated public population risk due to chemical separation accidents is 0.34 LCF. These risks must be combined with the risks of receiving/unloading and temporarily storing the foreign research reactor spent nuclear fuel, which were presented in Table 4-24. Assuming the foreign research reactor spent nuclear fuel would be received/unloaded and stored at RBOF for 13 years, the public MEI and population risks would be 0.0000026 LCF and 0.096 LCF, respectively.

The maximum of the two estimated accident-related MEI risks is 0.000047 LCF. This means that this hypothetical individual would have an additional chance of incurring an LCF of less than one in ten thousand.

The sum of the two population risks is 0.43 LCF. This means there would be an approximately 43 percent chance that one additional LCF would occur in the public population near the Savannah River Site due to accident conditions.

Idaho National Engineering Laboratory

Incident-Free Impacts at Idaho National Engineering Laboratory

The incident-free radiation dose rates for chemical separation at the Idaho National Engineering Laboratory are presented in Table 4-51 (DOE, 1995c). Multiplying these values by the estimated program duration of 12 years yields the doses presented in Table 4-52. These doses are converted into risks of LCF by applying the appropriate conversion factors and these results are also presented in Table 4-52.

Table 4-51 Incident-Free Radiation Dose Rates due to Chemical Separation at the Idaho National Engineering Laboratory

	Maximum Individual Dose Rate (mrem/yr)	Population Dose Rate (person-rem/yr)
Public		
Via Air	0.048	0.39
Via Water	0.0	0.0
Workers	5,000[a]	18

[a] *Assumed to be equal to the regulatory limit*

Table 4-52 Radiological Health Impacts Due to Incident-Free Chemical Separation Operations at the Idaho National Engineering Laboratory

	Maximum Individual Dose (mrem)	*Maximum Individual Risk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Public				
Via Air	0.58	2.9×10^{-7}	4.7	0.0024
Via Water	0.0	0.0	0.0	0.0
Workers	60,000	0.024	216	0.086

These risks must be combined with the risks of receiving/unloading the casks. Risks to the public were presented earlier in this chapter in Table 4-9. The risks of storage are also presented, but they are very low compared to those of receipt/unloading. Assuming the foreign research reactor spent nuclear fuel would be received at FAST for the full 13 years, the public MEI and population risks would be 2.5×10^{-9} LCF and 0.000021 LCF, respectively. These risks are much lower than the corresponding values in Table 4-52.

The handling-related risks to workers were presented earlier in this chapter. Under the conservative assumptions in the basic implementation of Management Alternative 1, the maximally exposed worker risk due to handling could be as high as 0.026 LCF which is higher than the 0.024 LCF in Table 4-52.

For the public, the estimated MEI risk due to incident-free chemical separation activities at the Idaho National Engineering Laboratory is less than one millionth of an LCF, which means that an individual who lives at the Idaho National Engineering Laboratory boundary would have an additional chance of less than one in a million of incurring an LCF.

The handling-related worker population risk at the Idaho National Engineering Laboratory is 0.10 LCF, from Table 4-15. This must be added to the 0.086 LCF from Table 4-52 to obtain the estimate of worker population risk due to incident-free chemical separation of foreign research reactor spent nuclear fuel, 0.19 LCF. This is near zero, so zero LCF would be expected among the radiation workers.

The estimated total public population risk at the Idaho National Engineering Laboratory is about 0.0024 LCF, which is much less than one LCF.

Impacts of Chemical Separation Accidents at the Idaho National Engineering Laboratory

DOE has analyzed the impacts of reasonably identifiable accidents due to chemical separation activities at the Idaho National Engineering Laboratory (DOE, 1995c). The accident with the highest estimated combination of frequency and consequence would be an inadvertent nuclear criticality during chemical separation. The accident frequency, consequences, and annual risks for this accident are presented in Table 4-53. Multiplying these results by the estimated program duration of 12 years yields the risks presented in Table 4-54.

Table 4-53 Annual Impacts of Chemical Separation Accidents at the Idaho National Engineering Laboratory

	Accident Frequency (per/yr)	*Consequences (LCF)*		*Risks (LCF/yr)*	
		Maximum Individual	*Population*	*Maximum Individual*	*Population*
Inadvertent Criticality					
Public	0.001	0.000025	0.0028	2.5×10^{-8}	0.0000028
Workers	0.001	0.0036	---	0.0000036	---

Table 4-54 Impacts of Accidents During Chemical Separation Operations at the Idaho National Engineering Laboratory

	Maximum Individual Risk (LCF)	*Population Risk (LCF)*
Public	3.0×10^{-7}	0.000034
Workers	0.000044	---

The highest estimated public MEI risk is 3.0×10^{-7} LCF, which means that an individual living at the management site boundary would have an additional chance of incurring an LCF of less than one in a million.

The highest estimated public population risk is 0.000034 LCF, which is much less than one LCF.

4.3.6.6.5 Waste Management

Savannah River Site

DOE has analyzed the wastes that would be generated from the aluminum-based foreign research reactor spent nuclear fuel and from an additional inventory of aluminum-based spent nuclear fuel during chemical separation activities. High-level waste, saltstone, transuranic waste, hazardous/mixed waste, and low-level waste would be generated under this implementation subalternative. All these wastes would be managed along with similar wastes at the Savannah River Site. The alternatives for managing all DOE wastes have been evaluated in the Draft Waste Management Programmatic EIS (DOE, 1995b).

The estimates of waste volumes that would be generated under this implementation subalternative are based on comparisons with similar operations on similar spent nuclear fuels. These aluminum-based spent nuclear fuel elements are similar to DOE's Mark 16/22 spent nuclear fuel elements at the Savannah River Site.

High-level liquid waste would be transferred to the F/H-Area Tank Farm for volume reduction and then to the Defense Waste Processing Facility for conversion into a borosilicate glass form suitable for prolonged storage. The high-level glass waste that would result from chemically separating the 18.2 MTHM (approximately 17,800 elements) of foreign research reactor spent nuclear fuel in this implementation subalternative would fill about 72 canisters (Dupont, 1996). Scaling this result up to include the total inventory of 51 MTHM yields an estimate of about 200 canisters. These canisters would be managed with the estimated 5,717 canisters that the Savannah River Site expects to produce from the existing onsite inventory of liquid high-level waste (WSRC, 1995). Each canister will contain approximately 40,000 curies of radioactivity (DOE, 1994a). The representative radionuclide composition of the waste glass is presented in Table 2.11 of the Integrated Data Base Report-1993 (DOE, 1994a). The radionuclides that contribute most of the radioactivity would be cesium-137, strontium-90, and their daughters. DOE expects that this waste form would be acceptable for disposal in a geologic repository.

Saltstone would be produced during the vitrification of high-level waste at the Defense Waste Processing Facility. An estimated 4,000 m^3 (140,000 ft^3) would be generated from the 18.2 MTHM of foreign research reactor spent nuclear fuel and would be disposed of onsite (Dupont, 1996). Scaling this result up to include the total inventory of 51 MTHM yields an estimate of about 11,300 m^3 (400,000 ft^3). This is much less than the maximum estimated cumulative saltstone to be generated at the Savannah River Site during the 10-year period from 1995 through 2004, which would be 625,211 m^3 (about 22,000,000 ft^3) (DOE, 1994b). The saltstone would contain far less radioactivity than the high-level waste glass: approximately 0.1 curie per cubic meter (DOE, 1994a). The approximate composition of the saltstone in terms of specific radionuclides is presented in Table C.5 of the Integrated Data Base Report-1993 (DOE, 1994a). The radionuclides that contribute most of the radioactivity would be promethium-147 until about 2000, then strontium-90 and its daughter thereafter.

Transuranic waste would not be generated during the chemical separation activities of foreign research reactor spent nuclear fuel (DOE, 1995a). The trace amounts of transuranic elements would not be removed from the waste stream, so they would be included in the high-level waste. If the Taiwan Research Reactor spent nuclear fuel (included in the total inventory of 51 MTHM) is chemically separated and the transuranic elements removed, then an estimated 832 m^3 (about 29,400 ft^3) of transuranic waste would be generated (DOE, 1995a). This is much less than the maximum estimated cumulative transuranic waste to be generated at the Savannah River Site during the 10-year period from 1995 through 2004, which would be 9,426 m^3 (about 333,000 ft^3) (DOE, 1994b).

Hazardous/mixed waste would also be produced under this implementation subalternative. An estimated 104 m^3 (about 3,700 ft^3) would be generated during 13 years of chemical separation operations (DOE, 1995a). This is much less than the maximum estimated cumulative mixed waste to be generated throughout the entire Savannah River Site during the 10-year period from 1995 through 2004, which would be 14,720 m^3 (about 520,000 ft^3) (DOE, 1994b).

Solid low-level waste would also be produced under this implementation subalternative. An estimated 74,000 m^3 (about 2,600,000 ft^3) would be generated during 13 years of chemical separation operations (DOE, 1995a) and would be disposed of onsite. This is much less than the maximum estimated cumulative low-level waste to be generated throughout the entire Savannah River Site during the 10-year period from 1995 through 2004, which would be 397,177 m^3 (about 14,000,000 ft^3) (DOE, 1994b).

Idaho National Engineering Laboratory

DOE has also analyzed the wastes that would be generated from the foreign research reactor spent nuclear fuel and from an additional inventory of spent nuclear fuel during chemical separations activities at the Idaho National Engineering Laboratory. High-level waste, low-level grout, transuranic waste, hazardous/mixed waste, and low-level waste would be generated under this implementation subalternative. All these wastes would be managed along with similar wastes at the Idaho National Engineering Laboratory. The alternatives for managing all DOE wastes have been evaluated in the Draft Waste Management Programmatic EIS (DOE, 1995b).

The estimates of waste volumes that would be generated under this implementation subalternative are based on comparisons with similar operations on similar spent nuclear fuels. For 12 MTHM of aluminum-based foreign research reactor spent nuclear fuel, 56 canisters of high-level waste glass would be generated (Denney, 1995). Scaling up to the 19.2 MTHM of foreign research reactor spent nuclear fuel yields 90 canisters. Scaling up further to the total inventory of 65 MTHM yields an estimate of about 300 canisters. These canisters would be managed along with the estimated 8,500 canisters Idaho National Engineering Laboratory expects to produce from the existing inventory of high-level waste onsite (DOE 1995b). Although the waste form has not been determined yet, each canister is estimated to contain approximately 22,000 curies of radioactivity (DOE, 1994a). The composition of the waste form in terms of specific radionuclides has not been determined yet, but it is reasonable to expect it to be similar to that of the glass at the Savannah River Site. DOE expects that the waste form would be acceptable for disposal in a geologic repository.

Another possibility exists if large quantities of nonaluminum-based spent nuclear fuels are being chemically separated in these facilities. Some aluminum is necessary to produce the stable waste form, and the 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel could satisfy this requirement. In this case, the chemical separation of the aluminum-based spent nuclear fuel would not increase the number of canisters that would be generated at the Idaho National Engineering Laboratory.

The estimates of low-level waste grout that would be generated under this implementation subalternative are also based on comparisons with similar operations on similar spent nuclear fuels. For 12 MTHM of aluminum-based foreign research reactor spent nuclear fuel, 1,280 m^3 (about 45,000 ft^3) of low-level waste grout would be generated (Denney, 1995). Scaling up to the 19.2 MTHM of foreign research reactor spent nuclear fuel in this implementation subalternative yields about 2,000 m^3 (70,629 ft^3). Scaling up further to the total inventory of 65 MTHM yields an estimate of about 6,900 m^3 (about 245,000 ft^3). This grout would be managed along with the other grout the Idaho National Engineering Laboratory would produce onsite. The grout is expected to contain far less radioactivity than the high-level waste glass/ceramic: much less than one curie per cubic meter. The composition of the grout in terms of all the specific radionuclides has not been determined yet, but the major radioactive constituents would be cesium-137 and strontium-90. The cesium-137 and strontium-90 concentrations in the grout are expected to be about 0.034 and 0.0093 curies per cubic meter, respectively (Bendixsen, 1995).

Transuranic waste would not be generated during chemical separation of the foreign research reactor spent nuclear fuel. Furthermore, the Idaho National Engineering Laboratory would not separate the transuranic elements from the Taiwan Research Reactor spent nuclear fuel if it were transported there from the Savannah River Site. Therefore, no transuranic waste would be generated during chemical separation of the additional inventory of spent nuclear fuel. The estimated amount of cumulative transuranic waste for 10 years with minimum waste management at the Idaho National Engineering Laboratory is 67,000 m^3 (about 2,400,000 ft^3) (DOE, 1995c).

Hazardous/mixed waste would also be produced under this implementation subalternative. Assuming a waste generation rate about equal to the rate at the Savannah River Site, an estimated 96 m^3 (3,400 ft^3) would be generated during 12 years of chemical separation operations. This is much less than the estimated 29,000 m^3 (1,020,000 ft^3) of cumulative hazardous and mixed waste to be generated throughout the entire Idaho National Engineering Laboratory during the next 10 years with minimum waste management (DOE, 1995c).

Solid low-level waste would also be produced under this implementation subalternative. Assuming a waste generation rate about equal to the rate at the Savannah River Site, an estimated 68,300 m^3 (2,400,000 ft^3) would be generated during 12 years of chemical separation operations and would be disposed of onsite. This is more than the estimated 47,000 m^3 (1,660,000 ft^3) of low-level waste to be generated throughout the entire Idaho National Engineering Laboratory during the next 10 years with minimum waste management (DOE, 1995c). The Idaho National Engineering Laboratory would treat the waste at the Waste Experimental Reduction Facility and send it to the Radioactive Waste Management Complex for onsite disposal.

4.3.6.7 Summary of the Impacts of Implementation Alternative 6 (Near Term Conventional Chemical Separation)

The principal impacts under this implementation alternative would be occupational and public health and safety impacts. These are presented in Table 4-55 in terms of the risk of death due to cancer during each of the four segments of this implementation alternative. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. The marine transport, port activities, and ground transport impacts are identical to the basic implementation of Management Alternative 1. The management site activity impacts were derived by comparing, and summing as appropriate, the handling impacts of the basic implementation of Management Alternative 1 and the impacts of chemical separation dedicated to foreign research reactor spent nuclear fuel. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-55 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have approximately a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 0.0000043 LCF.

The highest estimated accident MEI risk is 0.000047 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

The population risks were calculated by summing the appropriate spent nuclear fuel handling risks from the basic implementation of Management Alternative 1 with the risks of chemical separation at each management site and selecting the largest value. For example, the incident-free worker population risk of 0.21 LCF is the largest sum of the risks from that estimated for spent nuclear fuel handling operations under Phase 1 of the basic implementation of Management Alternative 1 and the estimated risk due to chemical separation dedicated to foreign research reactor spent nuclear fuel at the Savannah River Site or the Idaho National Engineering Laboratory. The sum of the above risks at the Idaho National Engineering Laboratory is 0.19 LCF [0.10 LCF from Phase 1 of the basic implementation (Table 4-15) and 0.086 LCF from chemical separation], and the corresponding value at the Savannah River Site is 0.21 LCF [0.10 LCF from Phase 1 of the basic implementation (Table 4-14) and 0.11 LCF from chemical separation].

As shown in Table 4-55, the total incident-free population risk would be 0.39 LCF for the potentially exposed public, while the corresponding risk would be 0.32 LCF for workers. Thus, there would be an estimated 39 percent chance of incurring 1 additional LCF among the exposed general public, and a 32 percent chance of incurring 1 additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-55. DOE and the Department of State estimate there could be about a 13 percent chance that a truck driver or member of the public could die in a traffic accident associated with this implementation alternative. This death would be unrelated to the radioactive nature of the cargo.

Table 4-55 Maximum Estimated Radiological Health Impacts of Implementation Alternative 6 (Near Term Conventional Chemical Separation)

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	Population	
		General Public	Workers
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 0.000029	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2×10^{-10}	0.000029	---
Ground Transport			
Incident-Free	0.00052	0.21	0.065
Accidents	1.3×10^{-11}	0.00014	---
Site Activities			
Incident-Free	0.026	0.18	0.21
Accidents	0.000047	0.43	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000047	----	----
Total Population Risk			
Incident-Free	----	0.39	0.32
Accidents	----	0.43	----

4.3.7 Implementation Alternative 7: New Developmental Treatment and/or Packaging Technologies

The environmental impacts of the developmental treatment and/or packaging technologies cannot be estimated with confidence at this time because the technologies and procedures are still under development. Implementation of certain of these technologies would require new facilities and thus would generate all the impacts associated with construction. Appropriate NEPA documentation would be prepared to support a decision on implementation of a new technology. The developmental treatment and/or packaging technologies are described in Chapter 2, Section 2.2.2.7.

The date at which a new facility would be operational is highly uncertain. A fairly simple technology implemented in existing facilities could be operational by 2000. On the other hand, the technology development, NEPA analysis, facility construction, and startup could take about 15 years for a complex technology. Thus, DOE could choose to implement one of the accept-and-store alternatives, in parallel with this alternative to prepare the foreign research reactor spent nuclear fuel for disposal. This may be necessary because foreign research reactor spent nuclear fuel may not be accepted in a geologic repository without some form of chemical processing or treatment. The repository acceptance criteria will not be final until a repository has been licensed.

Any new facilities would be designed to meet modern standards. The new design would minimize air and water emissions and the public and worker radiation doses at least as well as existing facilities, so DOE and the Department of State expect these impacts would be somewhat lower than those presented above for the conventional chemical separation technologies.

Some rough quantitative estimates are possible on the number of canisters that would be produced by some of the developmental technologies for disposal. Table 4-56 compares these estimates to the number of canisters that would be generated by chemical separation. The estimates of numbers of canisters that would be generated by the developmental treatment and/or packaging technologies do not depend on which DOE site performs the treatment and/or packaging.

Table 4-56 Comparison of Geologic Disposal Canisters for Various Technologies

Technology	*Approximate Number of Canisters*
Conventional Chemical Separation	
at the Savannah River Site	72
at the Idaho National Engineering Laboratory	90
Developmental Packaging Technologies	
Direct Disposal in Small Packages	140
Can-in-Canister	240
Developmental Treatment Technologies	
Melt and Poison	25
Chop and Poison	25
Melt and Dilute	180
Dissolve and Poison	950
Chop and Dilute	4,900
Dissolve and Dilute	11,800

The can-in-canister concept was recently introduced (Leventhal and Lyman, 1995), but it could be possible to implement it quickly at the Savannah River Site. Most of the foreign research reactor spent nuclear fuel elements would fit in cans of approximately 10 cm diameter and 85 cm length. If all of the approximately 22,700 elements were placed in these cans, the total canned volume would be about 150 m^3. Using the can-in-canister technology, this volume of glass would be displaced from high-level waste canisters to be produced in the Defense Waste Processing Facility. Since each canister has an internal volume of 0.625 m^3, displacing 150 m^3 of glass would require the production of approximately 240 additional high-level waste glass canisters at the Defense Waste Processing Facility.

The rest of the estimates of numbers of canisters that would be generated by the developmental technologies are scaled from a study (WSRC, 1994a) of the disposition of 7.3 MTHM of aluminum-based spent nuclear fuel, up to the 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel. The melt and poison or chop and poison technologies could produce the fewest canisters, as low as 25 canisters. The consolidate and poison technology could produce the next lowest number of canisters (about 140) among the developmental technologies analyzed. The can-in-canister, melt and dilute, dissolve and poison, chop and dilute, and dissolve and dilute technologies would produce increasing numbers of canisters, in that order. The most canisters would be produced by the dissolve and dilute technology: over 11,000 canisters. This uncertainty in the number of canisters translates into a large uncertainty in the cost of disposal. Furthermore, it is not clear which, if any, of these waste forms would be acceptable in a geologic repository.

4.4 Management Alternative 2: Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

The basic implementation of Management Alternative 1 of the proposed action and the seven implementation alternatives to the basic implementation of Management Alternative 1 are all based on acceptance of foreign research reactor spent nuclear fuel into the United States. As discussed in Chapter 2, the two subalternatives under Management Alternative 2 facilitate overseas management of foreign research reactor spent nuclear fuel. This section discusses their policy considerations and environmental impacts. For convenience, the two subalternatives under Management Alternative 2 are defined briefly below:

1. Subalternative 1a - Overseas storage of the foreign research reactor spent nuclear fuel with U.S. technical and/or financial assistance, and

2. Subalternative 1b - Overseas reprocessing of the foreign research reactor spent nuclear fuel with U.S. nontechnical assistance.

Under these subalternatives, no foreign research reactor spent nuclear fuel would be accepted into the United States. The United States would negotiate some form of technical assistance and/or financial incentives in return for maintaining some measure of control over the spent nuclear fuel containing uranium enriched in the United States.

4.4.1 Subalternative 1a: Overseas Storage with U.S. Assistance

Policy Considerations

The foreign research reactor spent nuclear fuel could remain in interim storage overseas. The number of foreign research reactor spent nuclear fuel management sites involved would be greater and the quality of storage technology in some countries might be lower than if the basic implementation of Management Alternative 1, or one of its seven implementation alternatives, was adopted.

The cost of this subalternative might be greater than the cost of the basic implementation of Management Alternative 1 because it might not take advantage of economies of scale. To set up a secure area and a nuclear material handling infrastructure, purchase a storage cask, transfer the spent nuclear fuel to the cask, and maintain the secure area and nuclear infrastructure for 40 years would cost tens of millions of dollars. To repeat this in several dozen countries could potentially push the total cost up into the range of hundreds of millions of dollars. Furthermore, after incurring this expense, all of the U.S. origin HEU would still be located in foreign countries where a change in government could reverse any commitment to withhold the material from production of nuclear weapons.

This subalternative would be economically attractive only in countries that already have nuclear infrastructures. In these cases, the addition of the spent nuclear fuel from research reactors to existing spent nuclear fuel inventories in storage would involve only incremental costs without all the startup costs.

If the United States does not accept any near term foreign research reactor spent nuclear fuel shipments, provision of U.S. technical and/or financial assistance for the development of safe and secure storage capabilities would help to alleviate some of the problems posed by a lack of sufficient storage capacity. However, this subalternative presents several drawbacks from a nuclear weapons nonproliferation policy standpoint. The accumulation overseas of ever larger amounts of spent nuclear fuel containing HEU poses a risk that such weapons-usable material might be illicitly diverted to a weapons program. Although U.S. assistance in maintaining adequate physical security for foreign research reactor spent nuclear fuel repositories may lessen the potential for diversion, the proliferation risks would still be greater than under the basic implementation of Management Alternative 1. As the foreign research reactor spent nuclear fuel ages, it would become less radioactive and thus a more attractive target for illicit diversion.

For countries that will not allow the indefinite storage in their territories of increasing quantities of spent nuclear fuel, this subalternative is not a viable option. Under this scenario, reactor operators in these countries, in order to avoid shutting down, might be forced to consider storing their spent nuclear fuel in other countries, where safe and secure management and material accountancy problems could exist and the risk of illicit diversion could be a concern. For example, Austria was reportedly approached by commercial interests from Belarus with an offer to store spent nuclear fuel from the ASTRA reactor for hard currency. (Since the "Offsite Fuels Policy" for HEU spent nuclear fuel expired in 1988, the Austrian government has required that for fresh fuel to enter the country, an equivalent quantity of spent nuclear fuel must be shipped out of the country.) The offer, which was rejected in support of nuclear weapons nonproliferation policies, is indicative of the scenarios that may develop as pressure builds on reactor operators to close the back end of their nuclear fuel cycle.

Impacts

There would be no environmental impacts on U.S. territory for the duration of the interim period.

4.4.2 Subalternative 1b: Overseas Reprocessing with United States Non-Technical Assistance

4.4.2.1 Overview and Policy Considerations

Foreign research reactor spent nuclear fuel could be reprocessed in foreign facilities and the resulting high-level waste vitrified or cemented. No U.S. reprocessing technology would be used in this subalternative. The inventory and conditions for management of foreign research reactor spent nuclear fuel under Subalternative 1b are the same as those under basic implementation of Management Alternative 1. The amount of HEU that would be removed from international commerce is the same as under basic implementation of Management Alternative 1 [4.6 metric tons (5.1 tons)]. To be consistent with U.S. nuclear weapons nonproliferation policy, however, bilateral agreements would have to be established with one or more foreign governments before DOE and the Department of State could consider implementation of such a subalternative.

The advantages and disadvantages of the technology used for reprocessing overseas would be essentially the same as those described for chemical separation in the United States as discussed in Section 2.2.2.6.

There are four sites in Europe at which reprocessing is conducted for commercial customers: the Marcoule and La Hague sites in France, and the Dounreay and Sellafield sites in the United Kingdom. The companies that operate these sites are strictly regulated by their government agencies. The facilities at La Hague and Sellafield are dedicated to oxide spent nuclear fuel from commercial reactors and are not likely candidates for reprocessing the metallic foreign research reactor spent nuclear fuel. All four of these sites routinely release small quantities of radionuclides into the environment and produce radioactive wastes. For example, in 1993 the releases from the Dounreay facility to the North Sea included 2.7 Ci of total alpha activity, 220 Ci of beta activity excluding tritium, and 27 Ci of beta activity from tritium. These releases represented 13 percent, 7.2 percent, and 0.8 percent of the applicable regulatory limits (Jones et al., 1994). The radionuclides released into the atmosphere and into a river or sea would flow across international boundaries. These releases would cause a small, unmeasurable increase in world-wide natural background radiation levels. The transport of vitrified high-level waste away from the reprocessing facility would also produce environmental impacts on foreign territory and possibly in international waters.

Since the United States does not encourage the development of reprocessing capabilities overseas, DOE and the Department of State would only consider this subalternative in France or the United Kingdom where the capability already exists. Reprocessing would most likely take place (as it already has in several instances) at the Dounreay facility—the sole facility currently willing and able to reprocess foreign research reactor spent nuclear fuel. France's facility in Marcoule does reprocess spent nuclear fuel from French research reactors, but does not currently accept such spent nuclear fuel from other nations for reprocessing.

The British and French regulatory agencies require the customer to accept the wastes as a condition of reprocessing spent nuclear fuel, so this option would be unavailable to those countries lacking the technical or legal capability to store or dispose of high-level waste. Alternatively, the United States might consider accepting the wastes from reprocessing.

4.4.2.2 Waste Generation at the Foreign Reprocessing Site

Reprocessing the foreign research reactor spent nuclear fuel would produce two distinct streams: the uranium and the waste products.

For spent nuclear fuel containing HEU, the HEU would be blended down to LEU at the reprocessing facility. If the LEU were then shipped to the United States, the resulting environmental impacts would be no greater than for ordinary nonhazardous cargo because LEU produces such a small radiation dose rate.

The British and French have decades of experience in conditioning nuclear waste at their four reprocessing facilities. In recent years, they have greatly reduced the volumes of wastes that require disposal. Both nations use the same technology for vitrifying their high-level waste, and both nations produce the same size high-level waste glass canister: 0.15 m^3 (5.3 ft^3). These canisters of high-level waste glass are expected to be suitable for disposal in geologic repositories. As of September 1993, France and the United Kingdom had filled more than 2,100 and 350 canisters with high-level waste glass, respectively (Masson, et al., 1994).

As a general rule, European reprocessing and vitrification of about 8 to 10 MTHM of spent nuclear fuel would generate about 1 m^3 (35.3 ft^3) of high-level waste in glass form (UKAEA, 1994; Masson, et al., 1994). Thus, if all 19.2 MTHM of the foreign research reactor spent nuclear fuel were reprocessed and vitrified overseas, DOE and the Department of State estimate that the total volume of vitrified high-level waste would be only about 2.4 m^3 (85 ft^3). DOE and the Department of State estimate that the high-level waste from reprocessing all the foreign research reactor spent nuclear fuel would fill about 16 European-sized canisters. For reference, this volume of glass waste would fill four American-sized canisters.

4.4.2.3 Removal of Waste from the Reprocessing Site(s)

The British and French governments do not accept responsibility for ultimate disposal of the high-level waste glass canisters for foreign customers. Both nations require that disposal of the high-level waste glass canisters and any other wastes generated during reprocessing of their spent nuclear fuel, including low-level waste, be the responsibility of the nation(s) hosting the reactors. At the Dounreay Site, however, only small amounts of low-level waste have been generated during reprocessing of spent nuclear fuel from research reactors. Many nations with foreign research reactors, however, do not have any capabilities to accept the high-level waste glass canisters. The United States may accept the intact foreign research reactor spent nuclear fuel from these nations while simultaneously encouraging the nations which can accept the canisters to reprocess their foreign research reactor spent nuclear fuel under the conditions noted in Section 4.4.2.1. This would be a combination of the basic implementation of Management Alternative 1 and Subalternative 1b (overseas reprocessing) of Management Alternative 2.

As another option under this subalternative, if the host nations cannot accept this responsibility, the United States would commit to accept the high-level waste glass canisters. This could provide the incentive necessary to convince reactor operators to cooperate with the RERTR program and to use LEU in their reactors. Some nations may refuse to reprocess or require the United States to take title to the foreign research reactor spent nuclear fuel prior to reprocessing.

DOE and the Department of State could begin accepting canisters into the United States within the first 10 years, or DOE and the Department of State could specify that they be stored at the reprocessing facility for decades. If the canisters were accepted in the near term, they would most likely be stored at the Savannah River Site because this site has already built a new storage facility with a capacity of 2,286 canisters. If the canisters were stored overseas for decades, then they would be transported directly to the geologic repository.

Marcoule produces vitrified waste, similar to U.S. vitrified waste. In the United Kingdom on the other hand, as a result of a different regulatory structure, the wastes from reprocessing of research reactor spent nuclear fuel are classified as intermediate-level radioactive wastes. (In the United States, these same materials would be classified as high-level radioactive wastes.) In the United Kingdom, the intermediate-level wastes are mixed with a special cement and poured into steel drums, which can then be buried. This waste form is dissimilar to the vitrified borosilicate glass high-level waste form that is expected to be produced in the United States, and is incompatible with United States radioactive waste disposal standards. The government of the United Kingdom might allow an exchange of vitrified commercial waste from Sellafield for cemented waste from Dounreay, which might allow the United States to accept vitrified high-level waste from the United Kingdom.

Transportation of vitrified high-level waste must conform to U.S. Department of Transportation (49 CFR Part 173) and NRC (10 CFR 71) regulations. Under this option, the European-sized glass canisters would be transported in "Type B" casks, which provide a high degree of assurance that cask integrity will be maintained with essentially no loss of radioactive contents or serious impairment of the shielding capability provided by the cask, even in severe accidents. DOE has prepared initial designs for a defense high-level waste cask for truck transportation of the Savannah River Site high-level waste. As initially designed, the defense high-level waste cask uses a solid body concept to absorb energy during an accident and normal transportation conditions. To minimize the exposure to gamma radiation, shielding would be provided by a depleted uranium liner inside the cask body. (Gamma radiation is high-energy, short wavelength electromagnetic radiation with properties similar to x-rays.) The regulatory limit for radiation dose rate outside the cask is 10 mrem per hour at 2 m (6.6 ft) from the edge of the vehicle. Casks transported under this option are assumed to emit this level of radiation. Currently, however, no casks for shipping high-level waste canisters by truck or rail have been certified by the NRC.

Each of these "Type B" casks would be large enough to hold two European-sized glass canisters. Thus, the option of overseas reprocessing with acceptance of approximately 16 high-level waste glass canisters would require about 8 cask shipments into the United States (versus 721 cask shipments by sea and 116 by land under the basic implementation of Management Alternative 1). Vitrified high-level waste shipments would use the same East Coast port(s) identified in Chapter 2 for foreign research reactor spent nuclear fuel. The same procedures and representative overland routes analyzed for foreign research reactor spent nuclear fuel would apply to these shipments of vitrified high-level waste. The management site for these canisters would be the Savannah River Site. Alternatively, they might be transported directly to the candidate geologic repository at Yucca Mountain, NV.

Each of the eight casks is assumed to contain the waste products associated with one-eighth of the foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1.

Marine Transport Impacts

Risks under Subalternative 1b were assessed using the same methodology used to evaluate risks associated with the transport of the foreign research reactor spent nuclear fuel. The major differences in the analysis are the number of cask shipments and the isotopic content within each transportation cask.

Impacts of Incident-Free Marine Transport

As with the shipment of foreign research reactor spent nuclear fuel, the primary impact of incident-free marine shipping of vitrified waste would be upon the crews of the ships. Most of the assumptions used in the analysis of the crew exposure to the spent nuclear fuel (see Section 4.2.2.2) have been used to analyze the impact of the shipment of vitrified waste. The primary contribution to the crew dose would come from the daily cargo inspection activities. Three crew members have been modeled as performing the inspections and the same three crew members are assumed to perform this task for the entire voyage. For the purposes of this analysis it has been assumed that the vitrified waste would be transported on a chartered vessel, there would be no intermediate port calls, and the shipment would originate in Europe (either the United Kingdom or France.)

As in the spent nuclear fuel analysis, either two or eight casks are assumed to be on each single voyage. This assumption results in exposure to two radiation fields during all activities that bring crew members into the vicinity of the transportation casks. Should all the casks be shipped at once, this assumption is equivalent to assuming that this single voyage is made with two casks per hold in one vessel. The crew risk would be the same for this single voyage as for four voyages with two casks per vessel.

Results of the marine incident-free risk analysis are presented in Table 4-57. Due to the reduced number of cask shipments, compared to the approximately 721 marine cask shipments of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1, the risks to the crew would be approximately 2 orders of magnitude lower than those calculated in Section 4.2.2.2 for the basic implementation of Management Alternative 1. The doses to the crew, including the maximally exposed worker, would be well below the DOE and NRC limits for public exposure of 100 mrem per year. If, however, all the casks were shipped in 1 year (perhaps all on one ship), then the maximally exposed worker dose would exceed the limit of 100 mrem per year. In this case, new inspectors would be used to keep each individual's dose below the limit.

Table 4-57 Incident-Free Marine Transport Impacts (Subalternative 1b)

	Maximally Exposed Worker Dose (mrem)	*Maximally Exposed Worker Risk (LCF)*	*Population Dose to Crew (person-rem)*	*Population Crew Risk (LCF)*
Per voyage (2 casks)	53	0.000021	0.19	0.00007
Entire program	210	0.000084	0.74	0.00030

Impacts of Accidents During Marine Transport

If the ship carrying a cask of vitrified waste were to catch fire at sea and the cask was sufficiently damaged by fire to release its contents, members of the ship's crew near the fire would be exposed to the released radioactive material. Any resulting plume carrying radioactive particles would disperse over the ocean, where there is no human population. Therefore, the ship's crew would be the only people exposed to the released radioactive material. The number of ship's crew members is considerably smaller than the population modeled within a short distance of an accident that occurs in the port. Therefore, consequences of a shipboard accident resulting in the release of radioactive material in a plume would be covered by the consequences of the accidents considered in the port analysis. As discussed below, because the oceans are a very dilute system, effects on marine biota would not be discernible.

If a collision or other accident (e.g., loss of a cask over the side in a storm) occurred in which an intact cask fell overboard, the fact that the cask would be immersed would not necessarily result in a release of its contents. Spent nuclear fuel casks are designed to withstand at least a 15-m (50-ft) immersion, and it has been demonstrated that the cask seals will remain intact at much greater depths (IAEA, 1990). Spent nuclear fuel casks, damaged or undamaged, can be recovered from water up to 200 m (660 ft) deep: well beyond the range typical of coastal and port depths. (Recovery at great depths, e.g., more than 2,000 m or 6,600 ft, is possible, but would be costly). It is reasonable to believe that a cask would be recovered in any incident involving the immersion of a cask in waters up to 200 m (660 ft) in depth.

The Nuclear Energy Agency of the Organization for Economic Cooperation and Development, Paris, France, estimated the impacts of various accident scenarios involving shipment of reprocessed commercial spent nuclear fuel. The Nuclear Energy Agency estimated that a damaged and unrecovered cask of high-level waste in coastal waters would result in a peak individual human dose of 6.5 mrem per year per MTHM (NEA, 1988). Dose and exposure estimates that follow are based on the estimates generated in the Nuclear Energy Agency study and are modified to take into account the content of the casks based on the shipment of all material from the foreign research reactor spent nuclear fuel program in eight cask shipments.

In the most extreme situation, where the accident occurs in coastal waters, the spent nuclear fuel is not recovered, and the cask is damaged, the peak dose to an individual human is estimated to be 19 mrem per year. The individual is assumed to reside near the shore and ingest seafood (fish, mollusk, and seaweed) harvested from the area in the immediate vicinity of the vitrified waste transportation cask. (For an initially intact cask, the dose would be expected to be considerably lower, approximately 0.3 mrem per year.) Peak biota doses are estimated at 0.8 mrad per year for fish, 0.9 mrad per year for crustaceans, and 19 mrad per year for mollusks, if the cask were damaged and not retrieved from coastal waters. With cask retrieval, both the peak dose to an individual and the biotic impacts would be considerably smaller. The results for the loss and failure of a single cask are lower than the peak impacts for the loss and failure of a single spent nuclear fuel cask (see Section 4.2.1.3), principally due to the lower leach rate for vitrified waste (see Appendix C).

In deep waters, the radioactive constituents of the vitrified waste would be released slowly over time into the surrounding waters if the cask were not recovered. Some of the radioactive material would be removed from the water by adhesion to suspended sediments. Assuming a damaged cask of vitrified high-level waste were submerged on the deep ocean bottom, the peak human individual dose to an individual residing along the coast and ingesting seafood harvested from the general area in which the breached submerged cask is located is estimated to be 0.000015 mrem per year.

Humans would not be the principally exposed species in a deep ocean accident involving vitrified waste casks. Using the Nuclear Energy Agency estimates and assuming that the damaged waste cask lay on the ocean floor where it slowly released its radioactive inventory, the peak doses to biota residing on the ocean floor in or near the uppermost sediment layer would be 0.9 rad per year for fish, 1.2 rad per year for crustaceans, and 41 rad per year for mollusks (NEA, 1988).

Harmful effects of chronic irradiation have not been observed in natural aquatic populations at dose rates less than 365 rad per year (NCRP, 1991). At doses an order of magnitude below this, as would be the case in an accident involving the vitrified waste from the foreign research reactor spent nuclear fuel, it is unlikely that either a population of marine biota or individual members of that population would be harmed by the radiation resulting from a spent nuclear fuel accident. Furthermore, no chemical hazard would be expected from the release of the contents of the vitrified waste canisters into the open ocean.

Using the same accident probabilities used in the marine transport analysis of the basic implementation of Management Alternative 1, risk estimates were developed for this subalternative. The MEI risk due to the loss of a vitrified high-level waste cask in the ocean is very low for the shipment of up to eight casks. The highest estimated risk to a human would occur in the accident scenario in which a cask is sunk and not recovered from coastal waters. This scenario would result in an MEI risk on the order of 1×10^{-10} mrem per year, which corresponds to about 2.7×10^{-15} LCF. This means that the chance of the MEI incurring one LCF due to this subalternative would be about one in one quadrillion.

Port Activity Impacts

Impacts of Incident-Free Port Activities

As with the shipment of the foreign research reactor spent nuclear fuel, the primary impact of incident-free port activities required to unload the vitrified waste casks is upon the workers: port handlers, inspectors, and port staging personnel. Most of the assumptions used in the analysis of the port worker exposure to the foreign research reactor spent nuclear fuel (see Section 4.2.2.2) have been used to analyze the impact of the shipment of vitrified waste.

Results of the port activities' incident-free risk analysis are presented in Table 4-58. Due to the reduced number of cask shipments, compared to the approximately 721 marine cask shipments of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1, the risks to the port workers are approximately 2 orders of magnitude lower than those calculated in Section 4.2.2.2 for the basic implementation of Management Alternative 1. The doses to the crew, including the maximally exposed worker, are well below the DOE and NRC limits for public exposure of 100 mrem per year.

Impacts of Accidents During Port Activities

The methodology used to evaluate the accident consequences and risks associated with port accidents is identical to that used to assess these items for the shipment of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1 (Section 4.2.2.3). The MACCS code was used with site-specific population and meteorology data to determine the consequences of an accident. The inventory (radionuclide content) of the transportation casks was determined by combining the radionuclide content of all of the vitrified waste to be returned to the United States under this subalternative and equally dividing it among the eight casks. In this analysis it was assumed that the Canadian spent nuclear fuel, which was assumed to be sent to the United States via truck in the analysis documented in Section 4.2.2.3, would be sent to Europe and reprocessed. The vitrified waste from this spent nuclear fuel is included in this analysis.

Table 4-58 Incident-Free Port Activity Impacts (Subalternative 1b)

Impacts Per Shipment				
	Maximally Exposed Worker Dose (mrem)	***Maximally Exposed WorkerRisk (LCF)***	***Population Dose (person-rem)***	***Population Risk (LCF)***
Inspectors	1.3	5.2×10^{-7}	0.0053	0.0000021
Port Handlers	0.46	1.8×10^{-7}	0.0015	0.00000061
Port Staging Personnel	0.38	1.5×10^{-7}	0.0046	0.0000018
Maximum	**1.3**	$\mathbf{5.2 \times 10^{-7}}$	----	----
Total	----	----	**0.011**	**0.0000042**
Impacts for the Entire Subalternative 1b				
	Maximally Exposed WorkerDose (mrem)	***Maximally Exposed Worker Risk (LCF)***	***Population Dose to Workers (person-rem)***	***Population Risk to Workers (LCF)***
Inspectors	10	4.0×10^{-6}	0.04	0.000017
Port Handlers	4	1.6×10^{-6}	0.01	0.0000048
Port Staging Personnel	3	1.2×10^{-6}	0.04	0.000015
Maximum	**10**	$\mathbf{4.0 \times 10^{-6}}$	----	----
Total	----	----	**0.09**	**0.000036**

The amounts of material released from the glass in various accident scenarios, called release fractions, are based on information developed for accident analysis at the Savannah River Site (DOE, 1994k). These release fractions are the same for the three accident categories analyzed for the spent nuclear fuel port accidents (the accident categories included collisions and collisions followed by fires). Therefore, these three accident categories were combined to form a single category for this analysis. Accident probabilities were developed for this single accident category at both the dock and in the approach to the dock.

Since all of the vitrified waste would be transported to the United States from Europe, only East Coast ports were selected for port-specific analysis of the accident consequences. The port accident analysis was performed for three East Coast ports: Philadelphia, PA; Charleston, SC; and MOTSU in North Carolina. These three ports represent a wide range of port city populations. As in the port accident analysis discussed in Section 4.2.2.3, these ports are not necessarily the selected ports of entry for the vitrified waste. They are intended to be representative of the range of populations, and therefore consequences, associated with all of the potential ports of entry.

Results of this analysis are presented in Table 4-59. The consequences of an accident in port involving a cask of vitrified waste would be lower than for the category 5 and 6 accidents involving the foreign research reactor spent nuclear fuel casks. This is a result of the much lower release fractions associated with the vitrified waste compared to the release fractions for the metallic spent nuclear fuel. However, for the vitrified waste, category 4 accidents result in the release of the same amount of material from the vitrified waste as the category 5 and 6 accidents. (For foreign research reactor spent nuclear fuel, category 4 accidents result in much smaller consequences.) Because the frequency of this category of accidents is two orders of magnitude higher than that for category 5 and 6 accidents, the port accident risks per single-cask shipment are higher for vitrified waste than for foreign research reactor spent nuclear fuel. The port accident population risks would be about the same order of magnitude as those under the basic implementation of Management Alternative 1 because of these category 4 accidents.

Table 4-59 Port Accident Risks (Subalternative 1b)

Port	Risk per Single-Cask Shipment of Waste		Risk of the Entire Waste Acceptance Option	
	Population Dose (person-rem)	LCF	Population Dose (person-rem)	LCF
Philadelphia	0.006	0.000003	0.05	0.00002
Charleston	0.001	0.0000007	0.01	0.000005
MOTSU	0.0005	0.0000002	0.004	0.000002

The MEI doses calculated for these accidents have a rather small variance. The largest estimated MEI dose is 740 mrem. The largest probability of one LCF (given that the accident has occurred) was 0.00035. Combining these estimates with the probability of a severity category 4 accident per shipment and the number of shipments results in an MEI risk of 1.8×10^{-8} LCF.

Ground Transport Impacts

Under Subalternative 1b, DOE and the Department of State would transport eight casks of vitrified high-level waste overland from an East Coast port(s) to a candidate geologic repository (in Nevada for example). The shipments may go directly from the port(s) to the candidate geologic repository or they might go from the ports to the Savannah River Site for storage, then from the Savannah River Site to the candidate geologic repository. Results are displayed in Figures 4-18 and 4-19.

Impacts of Incident-Free Ground Transport (Ports to Repository)

Impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The dose rate near vehicles carrying vitrified waste was assumed to equal the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the vehicle. Incident-free transportation of vitrified high-level waste was estimated to result in total latent fatalities that ranged from 0.00023 to 0.0032 over the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The estimated number of radiation-related LCF for transportation workers ranged from 0.0001 to 0.0008. The estimated number of radiation-related LCF for the general population ranged from 0.00009 to 0.0024, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0001 to 0.0005. The impacts of transportation of vitrified waste canisters are described in more detail in Appendix E.

To estimate the maximally exposed ground transport worker risk, DOE and the Department of State assumed all the vitrified waste was transported during a 1-year period and one truck driver received his annual limit of 100 mrem during that year. This dose translates into a risk of 0.00005 LCF.

Impacts of Accidents During Ground Transport (Ports to Repository)

Cumulative transportation accident risks over the vitrified waste shipment program are estimated to range from 0.0000002 to 0.0000059 LCF from radiation and from 0.00003 to 0.0016 for traffic fatality, depending on the transportation mode and the port(s) selected.

Impacts of Incident-Free Ground Transport (Ports to the Savannah River Site to Repository)

Impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The dose rate from casks containing vitrified waste was assumed to equal the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the vehicle. The

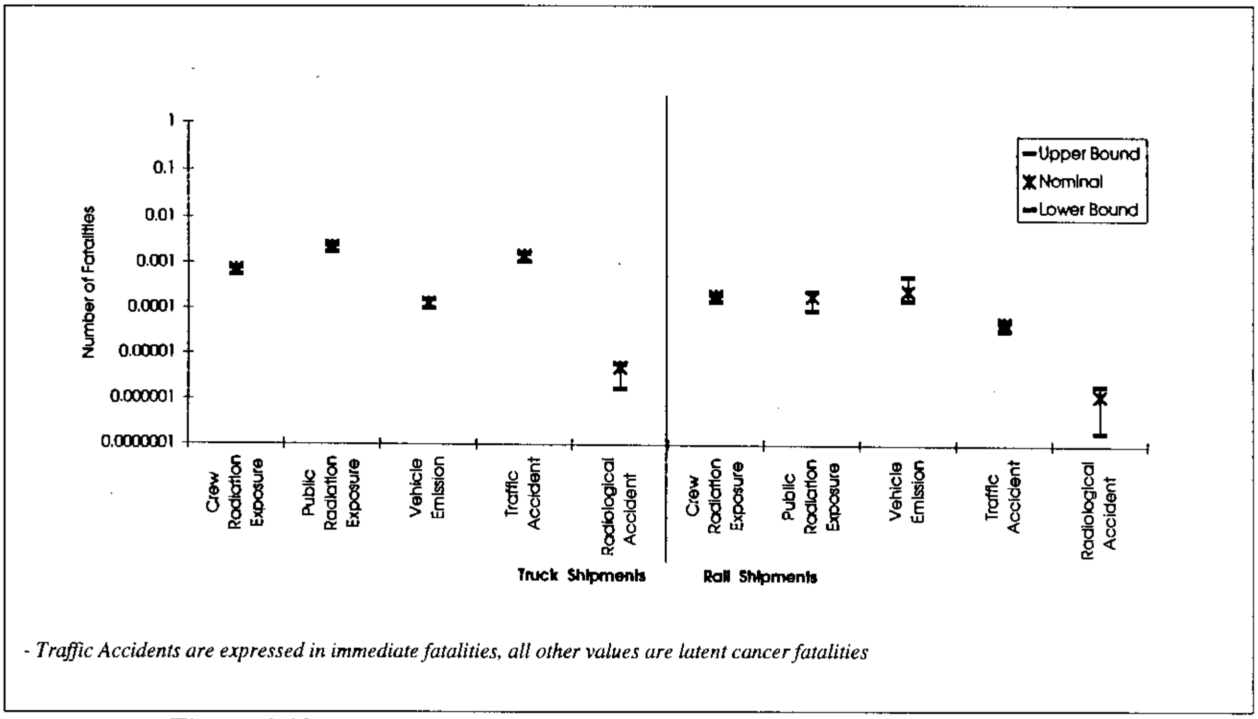


Figure 4-18 Range of Estimated Fatalities (Latent and Immediate) Under Management Alternative 2, Subalternative 1b (Ports to Repository)

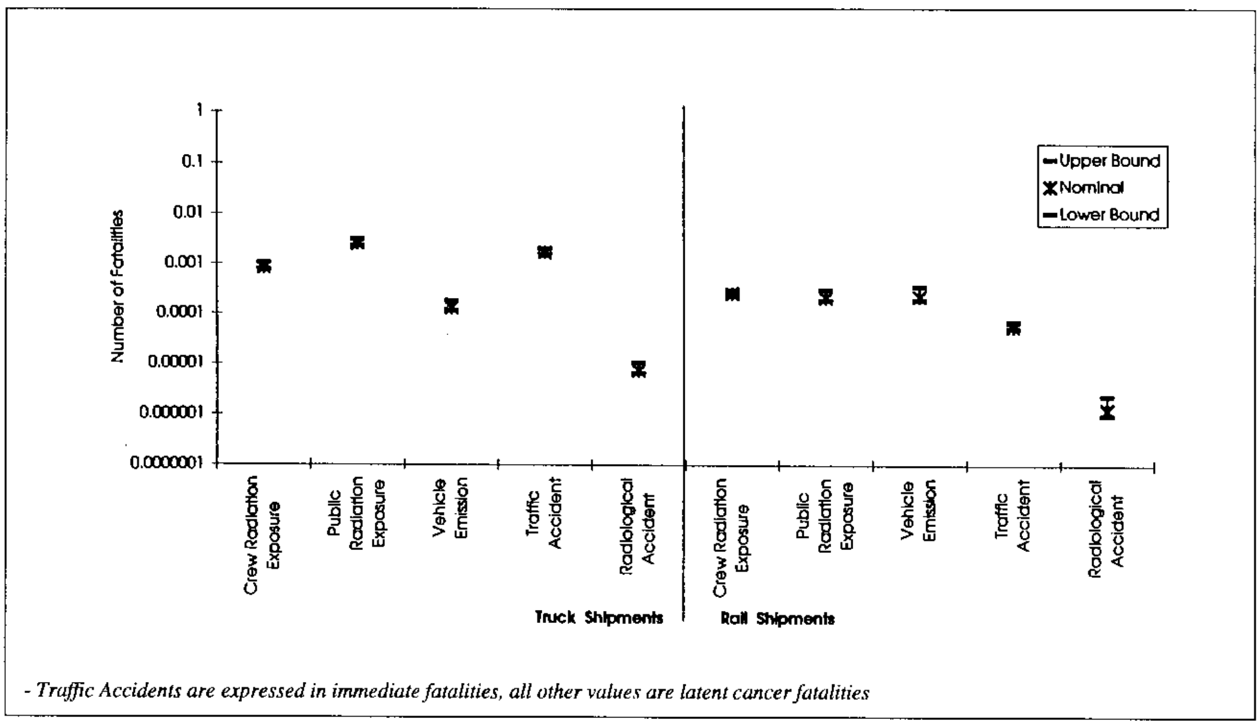


Figure 4-19 Range of Estimated Fatalities (Latent and Immediate) Under Management Alternative 2, Subalternative 1b (Ports to Savannah River Site to Repository)

incident-free transportation of the vitrified high-level waste was estimated to result in total latent fatalities that ranged from 0.00041 to 0.004 over the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The estimated number of radiation-related LCF for transportation workers ranged from 0.00023 to 0.001. The estimated number of radiation-related LCF for the general population ranged from 0.00018 to 0.003, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.00011 to 0.00035. Impacts of transportation of vitrified waste canisters are described in more detail in Appendix E.

To estimate the maximally exposed worker risk, it was assumed that the two legs of ground transport would be separated by a long storage period. That is, the second leg (transport from the Savannah River Site to the repository) would occur at least 20 years after the first leg (transport from the ports to the Savannah River Site). Thus, one individual truck driver would probably not be involved in both legs. DOE and the Department of State further assumed that each leg would last no more than 1 year, so no individual truck driver could receive more than the annual regulatory limit of 100 mrem. This translates into a maximally exposed worker risk of 0.00005 LCF.

Impacts of Accidents During Ground Transport (Ports to the Savannah River Site to Repository)

Cumulative transportation accident risks over the vitrified waste shipment program are estimated to range from 0.000001 to 0.00001 LCF from radiation and from 0.00005 to 0.002 for traffic fatality, depending on the transportation mode and the port(s) selected.

The consequences of the maximum foreseeable offsite transportation accident are greater than those of the basic implementation of Management Alternative 1. The frequency, however, is lower due to the reduced amount of ground transport. Maximum estimated MEI risk is reduced to 7×10^{-12} LCF.

Management Site Impacts

Impacts of Incident-Free Management Site Activities

Environmental impacts associated with the receipt and storage of the vitrified high-level waste canisters under Subalternative 1b are limited to the exposure of the working crew that would handle the incoming canisters at the site. The 16 canisters of vitrified waste (approximately 0.15 m^3 or 5.3 ft^3 each) would be received in 8 shipping casks and stored at the Glass Waste Storage Building at the Savannah River Site. The facility, described in Appendix F, has been designed for vitrified waste and has space for 2,286 canisters. Vitrification of all existing liquid high-level waste at the Savannah River Site is expected to produce a total of approximately 5,717 canisters. The impact of this additional amount of glass waste on the operational characteristics of the facility would be very low.

Vitrified waste would not contain any gaseous fission products, so there is no mechanism for incident-free emissions of radioactive material. Thus, impacts to the public near the Savannah River Site under this subalternative would be equal to zero.

To estimate the maximally exposed worker dose, DOE and the Department of State assumed that all the canisters would be received during one year. This is reasonable because of the small number of cask shipments. Then DOE and the Department of State conservatively assumed that one of the workers involved in handling these shipments would receive the maximum annual dose of 5,000 mrem allowed by regulation. This dose translates into an increased risk of 0.002 LCF.

The population dose to workers handling the eight casks would be 2.6 person-rem, based on the methodology presented in Appendix F, Section F.5 for unloading and storing in a vault-type dry storage structure. This translates into a worker population risk of 0.001 LCF.

Impacts of Accidents Onsite

The addition of 16 European-sized canisters to the thousands of larger American-sized canisters is expected to increase the accident risk by a very small increment, so this increase in the risk was not specifically analyzed in this EIS. The accident analysis for the Defense Waste Processing Facility has been reported in its Final EIS (DOE, 1994e).

Since vitrified waste contains no gaseous fission products, however, it is clear that the spent nuclear fuel element breach accident scenarios are not applicable to this subalternative. Thus, the aircraft-crash-with-fire scenario would present the highest risks. The highest annual estimates of MEI/NPAI and population risks under the basic implementation of Management Alternative 1 for this accident scenario are 1.2×10^{-9} LCF and 0.0000015 LCF, respectively (see Section 4.2.4.1). DOE and the Department of State consider these estimates to cover the risks for vitrified waste because the vitrified waste is designed to be much more stable than spent nuclear fuel in all accidents. Multiplying these annual estimates by the number of years the accident might occur (30 years) yields the risks for this alternative: 3.6×10^{-8} LCF for the MEI/NPAI risk and 0.000045 LCF for the population risk.

4.4.2.4 Disposal Site Impacts

Whether the vitrified high-level waste canisters were managed at the Savannah River Site or in Europe, eventually they would be transported to a geologic repository for disposal under this subalternative. Current planning for the U.S. candidate geologic repository at Yucca Mountain in Nevada indicates that acceptance of high-level waste canisters would begin early enough that the high-level waste from foreign research reactor spent nuclear fuel could be shipped to and emplaced in the repository before the end of the interim period.

Impacts due to handling European-sized canisters at the repository would be similar to the impacts due to handling American-sized canisters. After emplacement in the disposal site, no more impacts are expected to workers, the public, or the environment for at least 10,000 years because the radioactive material would be extremely unlikely to escape from the repository.

4.4.2.5 Summary of the Impacts of Subalternative 1b

The principal impacts under Subalternative 1b would be occupational and public health and safety impacts. These impacts would be due to the acceptance of vitrified high-level waste into the United States from Europe. (If no high-level waste were accepted, then there would be no impacts on U.S. territory.) These impacts are presented in Table 4-60 in terms of the risk of death due to cancer during each of the four segments of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-60 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of high-level waste producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing

Table 4-60 Maximum Estimated Radiological Health Impacts of Subalternative 1b

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.000084	0	0.0003
Accidents	2.7×10^{-15}	much less than 0.00002	---
Port Activities			
Incident-Free	0.000004	0	0.000036
Accidents	1.8×10^{-8}	0.00002	---
Ground Transport			
Incident-Free	0.00005	0.003	0.001
Accidents	7×10^{-12}	0.00001	---
Site Activities			
Incident-Free	0.002	0	0.001
Accidents	3.6×10^{-8}	0.000045	---
Highest Individual Risk			
Incident-Free	0.002	----	----
Accidents	3.6×10^{-8}	----	----
Total Population Risk			
Incident-Free	----	0.003	0.0027
Accidents	----	0.000075	----

people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) during the 1 year of high-level waste acceptance.

The highest estimated incident-free individual risk is 0.002 LCF, which would apply to an onsite radiation worker. This individual would have a one in five hundred chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally worker risk. DOE estimates this risk to be very nearly zero LCF.

The maximum estimated accident MEI risk is 3.6×10^{-8} LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten million. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

The total incident-free population risk for both the general public and workers would be much less than one LCF.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-60. There is about a 0.2 percent chance that a truck driver or member of the public could die in a traffic accident associated with this subalternative. This death would be unrelated to the radioactive nature of the cargo.

4.5 Management Alternative 3 - Combination of Elements from Management Alternatives 1 and 2 (Hybrid Alternative)

As discussed in Section 2.4, DOE and the Department of State could combine implementation elements from Management Alternatives 1 and 2. Analysis of this example Hybrid Alternative does not signify its preference over other possible Hybrid Alternatives.

Under this Hybrid Alternative, DOE and the Department of State would facilitate reprocessing of the foreign research reactor spent nuclear fuel at western European reprocessing facilities (i.e., Dounreay or Marcoule), as in Management Alternative 2. It is assumed that the foreign research reactor operators in countries that can accept the reprocessing waste would agree to this arrangement. DOE would accept and manage the remaining foreign research reactor spent nuclear fuel in the United States as in Management Alternative 1. (Refer to Section 2.4 for a more detailed description of this Hybrid Alternative).

Based on the current capabilities of overseas reprocessors, and for purposes of this analysis, only aluminum-based foreign research reactor spent nuclear fuel is assumed to be considered for reprocessing; all TRIGA spent nuclear fuel is assumed to be stored in the United States.

Under the Hybrid Alternative, the aluminum-based foreign research reactor spent nuclear fuel to be managed in the United States would be chemically separated at the Savannah River Site as in Implementation Alternative 6 to Management Alternative 1 (near term chemical separation in the United States), discussed in Sections 2.2.2.6 and 4.3.6. The uranium and waste products from this chemical separation would be managed as described in Sections 2.2.2.6 and 4.3.6, and the impacts of these activities would be covered by the impacts presented in those sections. The TRIGA spent nuclear fuel would be transported to the Idaho National Engineering Laboratory where it would be stored at existing storage facilities until ultimate disposition. This distribution of the spent nuclear fuel is consistent with the Programmatic SNF&INEL Final EIS (DOE, 1995c) Regionalization by Fuel Type alternative.

The environmental impacts associated with the foreign research reactor spent nuclear fuel that would be accepted into the United States, and the policy considerations of the Hybrid Alternative, are discussed below.

Policy Considerations

Under the Hybrid Alternative, up to 5.3 MTHM and about 5,600 elements of foreign research reactor spent nuclear fuel would be reprocessed overseas. The rest of the foreign research reactor spent nuclear fuel included in the basic implementation of Management Alternative 1, up to 13.9 MTHM and about 17,100 elements, would be accepted into the United States. Overall, the same amount of HEU as in the basic implementation of Management Alternative 1 would be removed from international commerce, up to about 4.6 metric tons (5.1 tons) of HEU.

4.5.1 Marine Transport Impacts

Impacts of Incident-Free Marine Transport

Impacts of incident-free marine transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. Incident-free transportation of spent nuclear fuel was estimated to result in total LCF that ranged from 0.021 to 0.024 over the 13-year duration of the acceptance program. These fatalities are the sum of the estimated number of radiation-related LCF to the ships' crews.

The range of impacts results from the analysis of shipment of the spent nuclear fuel on regularly scheduled commercial breakbulk vessels and on chartered container vessels and would be the same as for vessels analyzed under the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the difference between the two estimates is a result of the shorter vessel journey time for chartered vessels due to the intermediate port stops associated with the regularly scheduled commercial transport of the spent nuclear fuel.

The highest estimate of the incident-free maximally exposed worker risk is the same as for the basic implementation of Management Alternative 1 (0.00052 LCF for all the shipments combined).

Impacts of Accidents During Marine Transport

Population risks due to accidents under the Hybrid Alternative would be reduced from those associated with the basic implementation of Management Alternative 1 because of the reduced amount of marine transport. As before, the population risks of accidents at sea are bounded by the risk of accidents in port.

The maximum consequences of the at-sea accidents for the Hybrid Alternative are no different than those of at-sea accidents associated with the basic implementation of Management Alternative 1. For an accident involving the loss of a transportation cask in coastal waters, the maximum exposure to an individual is estimated to be 14,000 mrem per year. DOE and the Department of State would mitigate this impact, however, by recovering the cask. Due to the reduced number of cask shipments compared to the basic implementation of Management Alternative 1, the likelihood of such an accident would also be reduced. The Hybrid Alternative would require approximately 63 percent of the number of shipments required under the basic implementation of Management Alternative 1. The highest estimated risk due to an accident during marine transport would therefore be 0.00012 mrem per year peak dose to a human from the loss of a damaged cask in the deep ocean. This corresponds to an MEI risk of about 3×10^{-10} LCF. This means that this individual would have a chance of less than one in a billion of incurring an LCF due to an accident during marine transport.

4.5.2 Port Activity Impacts

Impacts of Incident-Free Port Activities

In the analysis of the basic implementation of Management Alternative 1, the radiological impact of port activities was estimated on a per-shipment basis. The Hybrid Alternative would require about 63 percent of the number of cask shipments required under the basic implementation of Management Alternative 1. The incident-free impacts of the port activities are proportionally reduced. The estimated number of LCF associated with this alternative range from 0.0021 to 0.0076. As in the marine incident-free analysis, this range of impacts is the result of the analysis of two modes of spent nuclear fuel shipment, regularly scheduled commercial breakbulk vessels and chartered container vessels.

The highest estimate of incident-free maximally exposed worker risk is the same as for the basic implementation of Management Alternative 1 (0.00052 LCF).

Impacts of Accidents During Port Activities

Port accident risks were calculated based on the per-shipment risks determined in the analysis of the basic implementation of Management Alternative 1. The analysis examined the impact of using a wide range of ports based on the population around the port city, from high density population ports such as Elizabeth, NJ, to low-density ports such as the MOTSU terminal in North Carolina. The analysis also considered the impact of chartered shipments (no intermediate port stops before the vessel reaches the spent nuclear fuel port of entry) versus regularly scheduled commercial shipments with up to two intermediate ports of call before the spent nuclear fuel port of entry. Port accident risks associated with the Hybrid Alternative are estimated to range from 2×10^{-7} to 0.00002 LCF from radiation. The range of fatality estimates is due to both the differences in port city populations and the number of intermediate port stops.

Consequences of the maximum foreseeable port accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced number of cask shipments, so the MEI risk is reduced to about 1 x 10^{-10} LCF.

4.5.3 Ground Transport Impacts

Impacts of Incident-Free Ground Transport

Radiological impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The results are presented in Figure 4-20. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.011 to 0.15 over the 13-year duration of the acceptance program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates is caused by two factors: the option of using truck or rail to transport spent nuclear fuel and the possibility of using different ports that created varying shipment distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.008 to 0.037. The estimated number of radiation-related LCF for the general population ranged from 0.010 to 0.11, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0031 to 0.025. Since these risk numbers are much less than one, implementation of the Hybrid Alternative would be unlikely to result in one LCF.

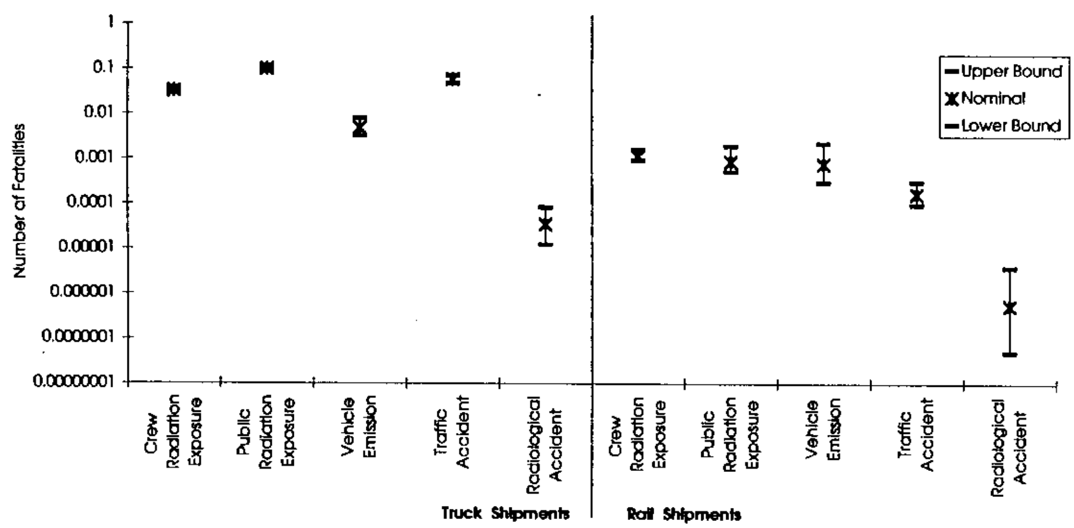


Figure 4-20 Range of Estimated Fatalities (Latent and Immediate) Under Management Alternative 3 (the Hybrid Alternative)

Impacts of Accidents During Ground Transport

Transportation accident population risks over the entire Hybrid Alternative are estimated to range from 0.000005 to 0.000081 LCF from radiation and from 0.002 to 0.069 for traffic fatality, depending on the transportation mode and the ports that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation.

The maximum foreseeable offsite transportation accident is identical to that for the basic implementation of Management Alternative 1. The risk is reduced to 7.1×10^{-12} LCF due to the reduced amount of ground transport.

4.5.4 Management Site Impacts

Under the Hybrid Alternative, the amount of foreign research reactor spent nuclear fuel that would be accepted into the United States is about 17,100 elements and 13.9 MTHM. All the TRIGA spent nuclear fuel, representing approximately 4,900 elements and 1.0 MTHM, would be received and stored in existing facilities at the Idaho National Engineering Laboratory. Aluminum-based spent nuclear fuel, representing approximately 12,200 elements and 12.9 MTHM, would be received and chemically separated at the Savannah River Site as described in Implementation Alternative 6 to Management Alternative 1 (near term chemical separation in the United States). Environmental impacts associated with the receipt and storage of the TRIGA spent nuclear fuel at existing facilities at the Idaho National Engineering Laboratory would be covered by the impacts presented for the basic implementation of Management Alternative 1 without construction of new facilities (Section 4.2). Environmental impacts associated with the receipt and chemical separation of the aluminum-based spent nuclear fuel at the Savannah River Site would be covered by the impacts presented for the near-term chemical separation alternative at the Savannah River Site (Section 4.3.6). The occupational and public health and safety impacts for both sites were estimated by combining the appropriate results from earlier analyses for the Idaho National Engineering Laboratory and the Savannah River Site.

Impacts to the Public of Incident-Free Management Site Activities

The approximately 4,900 elements that would be received and managed at the Idaho National Engineering Laboratory under this alternative represent about 22 percent of the number of elements that would be received and managed there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions from both aluminum-based and TRIGA foreign research reactor spent nuclear fuel during receipt and management at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 are presented in Table 4-9. Applying these results to the Hybrid Alternative at the Idaho National Engineering Laboratory for only TRIGA spent nuclear fuel is conservative because the TRIGA spent nuclear fuel would produce less gaseous fission product emissions than the mixture of spent nuclear fuel in the basic implementation of Management Alternative 1. Multiplying the results in Table 4-9 by the maximum duration of each activity (13 years for receipt and 40 years for storage) yields the highest estimated risks for this part of the Hybrid Alternative. The receipt/unloading impacts are reduced by the factor of 22 percent. The highest estimated public MEI risk is 7.8×10^{-10} LCF and the highest estimated public population risk is 0.0000064 LCF.

The approximately 12,200 elements that would be received at the Savannah River Site under this alternative represent about 54 percent of the number of elements that would be received and temporarily stored there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions during receipt at the Savannah River Site under the basic implementation of Management Alternative 1 are presented in Table 4-8. The impacts for storage in RBOF are much smaller than those for receipt. Multiplying these results by 54 percent and the maximum duration of 13 years yields the highest estimated risks for this part of the Hybrid Alternative. The highest estimated public MEI risk is 3.9×10^{-10} LCF and the corresponding estimated public population risk is 0.000020 LCF.

The approximately 12.9 MTHM that would be chemically separated at the Savannah River Site under this alternative represents about 71 percent of the MTHM that would be chemically separated there under Implementation Alternative 6 dedicated to foreign research reactor spent nuclear fuel. Public impacts due to this implementation alternative were presented earlier in this chapter in Table 4-48. Multiplying these results by 71 percent yields the estimated impacts to the public near the Savannah River Site due to this part of the Hybrid Alternative. Using this procedure, the highest estimated incident-free public MEI risk at the Savannah River Site is 0.0000031 LCF. The highest estimated incident-free public population risk at the Savannah River Site (including both the air and water exposure pathways) is 0.13 LCF.

The maximum of the three onsite activities' estimated public incident-free MEI risks is equal to 0.0000031 LCF, which would result from chemical separation activities at the Savannah River Site (The three parts are receipt and management of TRIGA spent nuclear fuel at the Idaho National Engineering Laboratory, receipt and temporary management of aluminum-based spent nuclear fuel at the Savannah River Site, and chemical separation at the Savannah River Site). Thus, the chance of this individual incurring an LCF due to the Hybrid Alternative would be less than one in one hundred thousand.

The total of the three onsite activities' estimated public incident-free population risks is 0.13 LCF.

Impacts to Workers of Incident-Free Management Site Activities

Incident-free maximally exposed worker radiation dose depends upon the duration of the receipts, not the amount of spent nuclear fuel involved. The duration of this Hybrid Alternative is 13 years, the same as that in both the basic implementation of Management Alternative 1 and Implementation Alternative 6. Thus, the estimated maximally exposed worker dose is also the same. The maximally exposed worker risk is estimated to be 0.026 LCF.

Incident-free worker population impacts due to the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory were presented in Section 4.2.4. Using the same evaluation process described in Appendix F, Section F.5, for the 162 casks of TRIGA foreign research reactor spent nuclear fuel that would be received and unloaded under this Hybrid Alternative yields a dose of 52 person-rem (dry storage in existing facilities). The associated worker population risk for this part of the Hybrid Alternative is 0.021 LCF.

Workers at the Savannah River Site would receive and unload 406 casks of aluminum-based foreign research reactor spent nuclear fuel in an existing wet facility under this alternative, receiving a population dose of 157 person-rem. The associated worker population risk for this part of the Hybrid Alternative is 0.063 LCF.

Incident-free worker population impacts due to Implementation Alternative 6 (chemical separation) were presented earlier in this chapter in Table 4-48. Multiplying these results by 71 percent yields the estimated incident-free impacts to the workers at the Savannah River Site due to the Hybrid Alternative. Using this procedure, the highest estimated incident-free worker population risk due to chemically separating this spent nuclear fuel at the Savannah River Site is 0.078 LCF.

The total of the three onsite activities' estimated incident-free worker population risks is 0.16 LCF.

Impacts of Accidents Onsite

Accident scenarios, frequencies, consequences, and annual risks for the Hybrid Alternative are derived from those for the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory and Implementation Alternative 6 at the Savannah River Site.

Annual accident risks for receipt, unloading, and storage at the Idaho National Engineering Laboratory were presented earlier in this chapter in Table 4-25. Multiplying these by the duration of the activity (13 years for receipt and 40 years for storage) yields the risk due to accidents at the Idaho National Engineering Laboratory under this alternative. The receipt/unloading impacts are reduced by the factor of 22 percent. These estimates are conservative because the TRIGA spent nuclear fuel involved would release fewer fission products than would the mixture of TRIGA and aluminum-based spent nuclear fuel in the basic implementation of Management Alternative 1. The highest estimated accident MEI risk for this part of the Hybrid Alternative is 1.9×10^{-6} LCF, which is due to an accidental criticality in a wet storage facility. The highest estimated accident population risk for this part of the Hybrid Alternative is 0.0088 LCF, which is due to the same accident scenario.

Annual accident risks for receipt and unloading at the Savannah River Site were presented in Table 4-24. Multiplying these by the duration of the receipt activity (13 years) yields the risk due to receipt and temporary storage accidents under this alternative. The highest estimated accident MEI risk for this part of the Hybrid Alternative is 2.6×10^{-6} LCF, which is due to an accidental criticality in RBOF. The corresponding estimated accident population risk for this part of the Hybrid Alternative is 0.096 LCF.

The accident MEI and population impacts due to chemical separation were presented earlier in this chapter in Table 4-50. Multiplying these results by 71 percent yields the estimated impacts at the Savannah River Site due to accidents under the Hybrid Alternative. Using this procedure, the highest estimated public MEI risk due to accidents during chemical separation at the Savannah River Site is 0.000033 LCF. The highest estimated accident population risk at the Savannah River Site is 0.24 LCF.

The maximum of the three onsite activities' estimated accident public MEI risks is equal to 0.000033 LCF, which would occur at the Savannah River Site. (The three onsite activities are receipt and management of TRIGA spent nuclear fuel at the Idaho National Engineering Laboratory, receipt and temporary management of aluminum-based spent nuclear fuel at the Savannah River Site, and chemical separation at the Savannah River Site). Thus, the chance of this individual incurring an LCF due to this Hybrid Alternative would be less than one in ten thousand.

The total of the three onsite activities' estimated accident population risks is 0.34 LCF.

4.5.5 Summary of the Impacts of the Hybrid Alternative

Principal impacts of the Hybrid Alternative would be occupational and public health and safety impacts. These are presented in Table 4-61 in terms of the risk of death due to cancer during each of the four segments of this alternative. The table also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population. In general, however, the implementation of the Hybrid Alternative would not pose higher risks than those determined for Management Alternative 1, assuming identical United States site management technology implementation. This is because the analyses in Management Alternative 1 aud its implementation alternatives considered the management of the maximum amount of foreign research reactor spent fuel in the United States.

Table 4-61 Maximum Estimated Radiological Health Impacts of the Hybrid Alternative

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.024
Accidents	3×10^{-10}	much less than 0.00002	---
Port Activities			
Incident-Free	0.00052	0	0.0076
Accidents	1×10^{-10}	0.00002	---
Ground Transport			
Incident-Free	0.00052	0.11	0.037
Accidents	7.1×10^{-12}	0.000081	---
Site Activities			
Incident-Free	0.026	0.13	0.16
Accidents	0.000033	0.34	---
Highest Individual Risk			
Incident-Free	0.026	----	----
Accidents	0.000033	----	----
Total Population Risk			
Incident-Free	----	0.24	0.23
Accidents	----	0.34	----

Table 4-61 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE management site receiving the maximum dose allowed by DOE regulation every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 7.8×10^{-8} LCF.

The highest estimated accident MEI risk is 0.000033 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-61, the total incident-free population risk would be 0.24 LCF for the potentially exposed public, while the corresponding risk would be 0.23 LCF for workers. Thus, there would be an estimated 24 percent chance of incurring one additional LCF among the exposed general public, and a 23 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-61. There is about a seven percent chance that a truck driver or member of the public could die in a traffic accident associated with this Hybrid Alternative. This death would be unrelated to the radioactive nature of the cargo.

4.6 No Action Alternative

Under the No Action Alternative, no foreign research reactor spent nuclear fuel or high-level waste would be accepted into or managed by the United States. The United States would not provide any technical or financial assistance to foreign research reactor operators for the management of their spent nuclear fuel. The United States would rely on the foreign governments' compliance with existing international agreements to control the disposition of foreign research reactor spent nuclear fuel containing uranium enriched in the United States.

Policy Considerations

The No Action Alternative would have a major adverse impact on U.S. nuclear weapons nonproliferation policy. The No Action Alternative would not remove any of the approximately 4.6 metric tons of U.S. origin HEU from international commerce as considered under the proposed action. Under this alternative, the foreign research reactor owners would continue, or may revert back to, use of HEU fuel in their reactors. Countries that can reprocess might send their HEU spent nuclear fuel to be reprocessed and use the separated HEU to produce fresh HEU fuel. In addition, any new research reactors to be built would likely be designed to use HEU fuel. Thus, the No Action Alternative could cause an increase in the number of shipments of weapons-grade nuclear material in transit around the world. It would also damage, perhaps irreparably, the credibility of the RERTR program. Countries that cannot reprocess their research reactor spent nuclear fuel would have to store their fuel. As the spent nuclear fuel ages, it becomes less dangerous to handle (its radioactivity decreases with time), and could possibly become a target of theft and diversion. Hence, the No Action Alternative would undermine the U.S. nuclear weapons nonproliferation policy and the risk of weapons-grade nuclear material being diverted into a nuclear weapons program would increase markedly.

To demonstrate the risk of having reactor owners continue, or revert back to, use of HEU fuel, please see Tables B-3, B-4, and B-5 in Appendix B. These tables list the 104 foreign research reactors whose spent nuclear fuel is included under the proposed action, including 24 reactors that have been converted (fully or partially) or are in various stages of conversion (i.e., ordered, or anticipated to begin converting) from HEU to LEU fuel, and 30 reactors that could be converted, but are not being converted, because the owners of the research reactors are awaiting the outcome of this EIS before they make a decision. Under the No Action Alternative, it is possible that up to 48 foreign research reactor operators could choose to continue or revert back to using HEU fuel in their reactors. These tables also list 23 foreign research reactor operators who possess HEU spent nuclear fuel, even though their reactors are either already shut down or planned to be shutdown for various reasons. This HEU spent nuclear fuel would remain in the foreign research reactor host countries, if the No Action Alternative is selected.

On the other side of the ledger, the benefits obtained from research reactors, described briefly in Section 1.1 of the EIS, would be diminished. Since the No Action Alternative means no U.S. assistance to foreign research reactor operators for managing their spent nuclear fuel, additional research reactors may be forced to shut down, because of lack of funds and/or long term storage capabilities. DOE and the Department of State cannot estimate the number of reactors that would actually be shut down because this would depend on each country's regulations regarding spent nuclear fuel storage. Nevertheless, the medical, industrial and environmental services provided by the shutdown research reactors would be lost. For medical services in particular, foreign research reactors produce radioisotopes used in nuclear medicine in the United States (as discussed in Sections 1.1 and 4.3.1.3 of the EIS). If some of these reactors were forced to shut down, a shortage of medical radioisotopes could occur in the United States. Since the U.S. medical requirements for radioisotopes are not likely to decrease in the near future, alternative sources would have to be found. This could involve an increased level of activity at existing U.S. research reactors or construction of a new reactor in the United States to supply the needed medical radioisotopes, with all the potential environmental impacts of these actions.

Environmental Impacts of Overseas Storage without U.S. Assistance

The material could remain in interim storage overseas. The number of storage sites involved might be greater and the quality of storage technology in some countries might be lower than if the U.S. was involved. Under this option, there would be environmental impacts in foreign countries, but none on U.S. territory, unless some of the material was diverted into nuclear weapons production.

Environmental Impacts of Overseas Reprocessing without U.S. Assistance

The material could be reprocessed and the resulting high-level waste could be vitrified or cemented in foreign facilities. Transport of spent nuclear fuel from the reactors to these facilities and the reprocessing activities would produce environmental impacts in foreign countries, but none on U.S. territory, except possibly in cases where some of the material was diverted into nuclear weapons production.

Under this option, the United States would not accept any shipments of vitrified high-level waste. The transport of vitrified high-level waste back to the country of origin and its storage and disposal would produce environmental impacts in foreign countries, but none on U.S. territory. The separated HEU would be more vulnerable to diversion into nuclear weapons production, and the increased reliance on HEU for fuel would increase the number of opportunities for diversion of this weapons grade material.

4.6.1 Overseas Storage Without U.S. Assistance

The material could remain in interim storage overseas. The number of storage sites involved would be greater and the quality of storage technology in some countries might be lower than under the other alternatives. In addition, as the spent nuclear fuel gets older, it becomes less dangerous to handle (its radioactivity decreases with time), and could more easily become a target of theft and diversion.

4.6.2 Overseas Reprocessing Without U.S. Assistance

The material could be reprocessed and the resulting high-level waste could be vitrified or cemented in foreign facilities. Transport of spent nuclear fuel from the reactors to these facilities and the reprocessing activities would produce environmental impacts only in foreign nations.

Under this option, the United States would not accept any shipments of vitrified high-level waste. The transport of vitrified high-level waste back to the country of origin and its storage and disposal would produce environmental impacts only in foreign nations.

4.7 Preferred Alternative

As discussed in detail in Section 2.9, the preferred alternative is to accept and manage the foreign research reactor spent nuclear fuel and target material in the United States. Under this alternative, the aluminum-based foreign research reactor spent nuclear fuel and target material would be transported to and managed at the Savannah River Site. The TRIGA foreign research reactor spent nuclear fuel would be transported to and managed at the Idaho National Engineering Laboratory.

The policy considerations, marine transport impacts, port activities impacts, ground transport impacts, and management site impacts of the preferred alternative presented in this section are based on analysis performed for the basic implementation of Management Alternative 1 (Section 4.2), Implementation Alternative 1c (Section 4.3.1.3), Implementation Alternative 6 (Section 4.3.6), and Implementation Alternative 7 (Section 4.3.7).

4.7.1 Policy Considerations

The policy considerations for the preferred alternative are similar to those described in Section 4.2 for Management Alternative 1. A critical result of implementing this preferred alternative would be the continued viability and vitality of the Reduced Enrichment for Research and Test Reactors (RERTR) Program, which has the goal of minimizing and eventually eliminating the use of HEU in civil nuclear programs by providing foreign research reactor operators with a continued incentive to participate. Similarly, the successful development of alternative fuels for research reactors and the expansion of the program to Russia, the other Newly Independent States, China, South Africa, and other countries, and the establishment of a world-wide norm discouraging the use of HEU, is dependent on a United States' commitment to action such as that embodied in this preferred alternative.

Another crucial consideration associated with the preferred alternative is the *Treaty on the Non-Proliferation of Nuclear Weapons*. The parties to the Non-Proliferation Treaty met in May of 1995 and agreed to extend the treaty indefinitely and without conditions. One key to the success of the 1995 Non-Proliferation Treaty Conference was the ability of the United States to convince other Non-Proliferation Treaty parties that the nuclear weapons states had complied with their obligations under Article IV of the Non-Proliferation Treaty to assist the non-nuclear weapons states with peaceful applications of nuclear energy.

Although the Non-Proliferation Treaty was extended indefinitely, the parties also agreed to review the treaty every five years to ensure that all parties are in compliance. Any country which has been compelled to shut down its research reactors could accuse the United States of not having complied with its treaty obligations. This accusation, however ill-founded, could be made not only by the affected countries, but by any country opposed to the interests of the United States.

Including target material as part of the preferred alternative maximizes the amount of HEU to be removed from international commerce. This includes all the HEU in the basic implementation of Management Alternative 1 [4.6 metric tons (5.1 tons) of heavy metal containing HEU] and all the HEU in the target material in Implementation Subalternative 1c [0.2 metric tons (0.2 tons) of heavy metal containing HEU]. The total amount that would be removed from international commerce is up to 4.8 metric tons (5.3 tons) of heavy metal containing HEU.

DOE's preferred alternative allows for the use of chemical separation under certain circumstances, such as when alternative technologies present higher safety risks, are more costly or are unavailable. If chemical separation is used to process the foreign research reactor spent nuclear fuel, the HEU would be blended down during the separation process to a low-enriched form that is unsuitable for nuclear weapons purposes (the blenddown is also required because the F-Canyon cannot safely process HEU beyond initial dissolution). No plutonium would be separated. Instead, the plutonium would be left in the waste stream with the high-level radioactive chemical separation wastes. In addition, the waste would be handled using technologies that are intended to be used for substantially larger quantities of preexisting wastes (e.g., vitrification of high-level liquid radioactive wastes, grouting for low-level wastes, and incineration for some supernatant).

This potential method of handling the foreign research reactor spent nuclear fuel would be consistent with United States nonproliferation policy, despite the use of chemical separation, because (1) it would reduce the worldwide stockpiles of this nuclear weapons material; (2) no plutonium would be separated; and (3) the chemical separation would not be taking place for either nuclear weapons or nuclear power purposes.

DOE is aware that the inclusion of chemical separation within the preferred alternative could be interpreted by some nations, organizations, and persons as a signal of endorsement of the use of chemical separation as a routine method of waste management for spent nuclear fuel or a reversal of United States policy on chemical separation. This would not be an accurate interpretation. The United States policy regarding chemical separation was established in Presidential Decision Directive 13, and DOE and the Department of State have determined that this preferred alternative is consistent with that policy. The draft version of this EIS indicated that chemical separation is a non-preferred technology. This final preferred alternative includes provision for possible chemical separation. DOE maintains a presumption that spent nuclear fuel would not be chemically separated unless there is an imminent health and safety risk, or other programmatic conditions, that cannot be addressed during the time period when no feasible alternative to chemical separation is available. These considerations will be addressed by the independent study described in Section 2.9.

4.7.2 Marine Transport Impacts

The marine transport impacts of the preferred alternative would be similar to those of the basic implementation of Management Alternative 1, with the addition of the target material shipments. As discussed in Section 4.3.1.3 and Appendix B, Section B.1.5, target material would be prepared for transport by changing it into either oxide or calcine form, and both forms might be accepted at some time during the proposed policy period. Even though it requires less marine transport, the oxide form presents a higher radiological risk under accident conditions because its smaller particle size is more easily dispersed in air. Therefore, to be conservative, the analysis of marine and port radiological accidents is based on the assumption that all the target material would be shipped as an oxide. The rest of the marine and port target material transport analysis is based on the assumption of 15 cask shipments, which is the maximum number of marine target material casks. This represents an increase of approximately two percent over the 721 marine cask shipments in the basic implementation of Management Alternative 1.

Marine transport to the West Coast of the United States would be limited to a maximum of approximately 38 casks, which slightly decreases the total number of days the ships would be at sea. Furthermore, DOE would strive to minimize the number of shipments necessary by coordinating shipments from several reactors at a time (i.e., by placing multiple casks [up to 8] on a ship). DOE currently estimates that approximately 5 shipments through the Naval Weapons Station at Concord, California would be necessary.

Impacts of Incident-Free Marine Transport

The highest estimated maximally exposed worker risk due to foreign research reactor spent nuclear fuel is 0.00052 LCF, which is based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years (Table 4-2). This means that the chance of this hypothetical individual incurring a latent cancer due to the preferred alternative would be less than one in a thousand.

The highest estimated population risk for all of the ships' crews involved in the marine transport of foreign research reactor spent nuclear fuel is about 0.034 LCF, as discussed in Section 4.2.1.2.

Target material contains far less radioactivity than foreign research reactor spent nuclear fuel. Each transportation cask of target material would produce a radiation dose rate far below the rate that was assumed for the foreign research reactor spent nuclear fuel. Thus, the rounded-off results of the incident-free radiological risk calculations for the basic implementation of Management Alternative 1 are not affected by the addition of up to 15 marine casks of target material.

Impacts of Accidents During Marine Transport

The risks associated with accidents at sea are bounded by the risks of the same accidents in ports because humans in the vicinity of accidents at sea are much fewer in number than even the least populated port.

Marine Transport Cumulative Impacts and Mitigation Measures

The marine transport cumulative impacts and mitigation measures for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.1.4 and 4.2.1.5, respectively.

4.7.3 Port Activities Impacts

Although all of the candidate ports of entry presented in Section 3 are acceptable, based on the port selection criteria described in Appendix D, DOE would prefer to use military ports. All aluminum-based foreign research reactor spent nuclear fuel and target material from overseas would arrive at candidate ports on the East Coast of the United States, preferably the Naval Weapons Station at Charleston, South Carolina. Up to approximately 38 casks of TRIGA foreign research reactor spent nuclear fuel would arrive at candidate ports on the West Coast of the United States, preferably the Naval Weapons Station at Concord, California.

Impacts of Incident-Free Port Activities

As shown in Table 4-5, the highest maximally exposed worker risk is 0.00052 LCF, which is based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years. This means that the chance of this hypothetical individual incurring a latent cancer due to the preferred alternative would be less than one in a thousand.

The highest estimated population risk for port workers is about 0.012 LCF, as discussed in Section 4.2.2.3.

As discussed under *Impacts of Incident-Free Marine Transport* above, each transportation cask of target material would produce a radiation dose rate far below the rate that was assumed for the foreign research reactor spent nuclear fuel. Thus, the rounded-off results of the incident-free radiological risk calculations for the basic implementation of Management Alternative 1 are not affected by the addition of up to 15 cask shipments of target material.

Impacts of Accidents During Port Activities

The radiological risks due to port accidents were estimated in the same manner as for the basic implementation (Section 4.2.2.3) and Implementation Alternative 1c (Section 4.3.1.3) of Management Alternative 1. The highest estimated population risk for the entire preferred alternative program is 7.1×10^{-7} LCF. This risk estimate is lower than the earlier alternatives due to the use of military ports in the preferred alternative. These ports are located in areas of low population density, so the number of people potentially affected is much lower. The addition of target material causes a very small incremental increase (3×10^{-9} LCF) in the risk.

Port Activities Cumulative Impacts, Mitigation Measures, and Environmental Justice

The port activities cumulative impacts, mitigation measures, and environmental justice for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.2.4, 4.2.2.5, and 4.2.2.6, respectively.

4.7.4 Ground Transport Impacts

The ground transport impacts were calculated under the assumption that only military ports would be used. DOE has selected military ports close to the management sites (the Charleston NWS in South Carolina and the Concord NWS in California) as the preferred ports of entry.

The risk estimates were maximized by assuming all target material would be oxide for radiological accident calculations and calcine for all other calculations. The calcine form could require up to 125 casks of target material to be shipped overland from Canada.

The preferred points of entry, destinations, and approximate numbers of cask shipments in the preferred alternative are presented in Table 4-62. Other shipment distributions would also be possible.

Table 4-62 Points of Entry, Destinations, and Numbers of Shipments in the Preferred Alternative

Cargo and Destination	*Point of Entry*			*Total Cask Shipments*
	East Coast	*West Coast*	*Canadian Border*	
Aluminum-Based Foreign Research Reactor Spent Nuclear Fuel to the Savannah River Site	559	0	116	675
TRIGA Foreign Research Reactor Spent Nuclear Fuel to the Idaho National Engineering Laboratory	124	38	0	162
Target Material to the Savannah River Site	up to 15	0	up to 125	up to 140
Total Cask Shipments	**up to 698**	**38**	**up to 241**	**up to 977**

Impacts of Incident-Free Ground Transport

The incident-free ground transport of foreign research reactor spent nuclear fuel and target material is estimated to result in a maximum of 0.089 LCF over the entire duration of the program. This is the sum of the estimated number of radiation-related LCF to the public and transportation workers.

The estimated maximum number of radiation-related LCF for transportation workers is 0.022. The estimated maximum number of radiation-related LCF for the general public is 0.067, and the estimated maximum number of non-radiation-related fatalities from vehicular emissions is 0.018.

Impacts of Accidents During Ground Transport

The total ground transport accident population risks for the preferred alternative are estimated to be less than 0.00072 LCF from radiation and 0.052 from traffic collisions.

The maximum foreseeable offsite transportation accident would involve a transportation cask of oxide target material in a suburban population zone under neutral (average) weather conditions, which could expose the MEI to 150 mrem. A similar event involving a transportation cask of spent nuclear fuel could expose the MEI to 2.4 mrem. These events are both in the highest accident severity category. Taking all the possible consequences and frequencies of these accidents into account, and adding the foreign research reactor spent nuclear fuel risks with the target material risks yields the MEI risk of 2.7×10^{-11} LCF for the preferred alternative.

Ground Transport Cumulative Impacts, Mitigation Measures, and Environmental Justice

The ground transport cumulative impacts, mitigation measures, and environmental justice for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.3.4, 4.2.3.5, and 4.2.3.7, respectively.

4.7.5 Management Site Impacts

As discussed in Section 2.9, all the TRIGA foreign research reactor spent nuclear fuel would be managed at the Idaho National Engineering Laboratory. The fuel would be received and stored in existing facilities. The environmental impacts of the preferred alternative at the Idaho National Engineering Laboratory can be estimated from the environmental impact analysis presented for the basic implementation of Management Alternative 1 (Section 4.2).

At the Savannah River Site, however, the impacts would vary depending on the specific outcome of the preferred management strategy at the site. Aluminum-based foreign research reactor spent nuclear fuel and target material would be managed at the Savannah River Site. The management of the foreign research reactor spent nuclear fuel is based on the schedule for successful implementation of a new treatment and/or packaging technology. If such a new technology could not be successfully demonstrated by the year 2000, chemical separation of a portion of the foreign research reactor spent nuclear fuel might be implemented. The foreign research reactor spent nuclear fuel and target material that is not chemically separated would be stored in existing facilities at the Savannah River Site until the new technology is operational.

Since the preferred alternative includes the construction and operation of an unspecified treatment and/or packaging technology at the Savannah River Site, the environmental impacts of this alternative at this site cannot be estimated with precision. DOE expects, however, that the radiological and nonradiological health and environmental effects from the construction and operation of facilities that would support a new technology would not exceed those estimated for construction of new dry storage facilities and operation of a conventional chemical separation facility evaluated in Sections 4.2.4.2 and 4.3.6 of this EIS. This expectation is based on the following general principles:

- New facilities would be constructed using current DOE design criteria which have evolved on the basis of increased protection of the public, workers, and the environment.
- The primary source of radiological releases from the chemical separation process is the front end dissolution of the spent nuclear fuel matrix. None of the new technologies considered involves a process that would produce greater releases.
- One of the reasons for the development of a new treatment and/or packaging technology is to reduce the volume and toxic nature of low-level and hazardous waste streams, an issue considered to be a disadvantage of the chemical separation process.

Nonradiological impacts from the construction of facilities that would support the new technology are expected to be typical to those assessed for the construction of new staging and storage facilities assessed for the basic implementation of Management Alternative 1 in Section 4.2.4.2. These include land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air and water quality, ecology, noise, materials and energy consumption, and non-radiological or non-toxic waste production during construction.

The occupational and public health and safety, waste management, and cumulative impacts presented below assume that the implementation of the preferred alternative at the Savannah River Site would result in radiological health effects equal to those presented in Sections 4.3.6 and 4.3.7 of this EIS.

4.7.5.1 Occupational and Public Health and Safety

Impacts to the Public of Incident-Free Management Site Activities

The approximately 4,900 foreign research reactor spent nuclear fuel elements that would be received and managed at the Idaho National Engineering Laboratory under the preferred alternative represent about 22 percent of the total number of elements that would be received and managed there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions from both aluminum-based and TRIGA foreign research reactor spent nuclear fuel during receipt and management at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 are presented in Table 4-9. Applying these results to the preferred alternative at the Idaho National Engineering Laboratory for only TRIGA spent nuclear fuel is conservative because the TRIGA spent nuclear fuel would produce less gaseous fission product emissions than the mixture of spent nuclear fuel in the basic implementation of Management Alternative 1. Adjusting the figures from Table 4-9 to account for the reduced amount of material in the preferred alternative yields the highest estimated risks for this part of the preferred alternative. The highest estimated public MEI risk is 7.8×10^{-10} LCF and the highest estimated public population risk is 0.0000064 LCF.

Radioactive emissions would not be expected from the target material receipt or storage because this material contains no gaseous fission products. Therefore, the incident-free radiological impacts to the public would be zero.

The incident-free radiological public health impacts at the Savannah River Site due to the preferred alternative are assumed to be equal to those discussed in Section 4.3.6.6.4 under the subheading, *Incident-Free Impacts at the Savannah River Site.* The highest estimated public MEI risk is 0.0000043 LCF and the highest estimated public population risk is 0.18 LCF.

The maximum of the onsite activities' estimated public incident-free MEI risks is equal to 0.0000043 LCF, which would occur at the Savannah River Site. The chance of this hypothetical individual incurring an LCF due to the preferred alternative would be less than one in one hundred thousand.

The total of the onsite activities' estimated incident-free population risks to the people who live near both sites is equal to 0.18 LCF. This number means that there would be an approximately 18 percent chance of one additional LCF among the population residing around the two sites due to these incident-free activities.

Impacts to Workers of Incident-Free Management Site Activities

Incident-free maximally exposed worker radiation dose depends upon the duration of the receipts, not the amount of spent nuclear fuel involved. The duration of the receipts in the preferred alternative is 13 years, the same as that in the basic implementation, the target material alternative, and the chemical separation alternative of Management Alternative 1. Thus, the estimated maximally exposed worker dose is also the same. The highest maximally exposed worker risk is estimated to be 0.026 LCF.

The incident-free worker population risks of the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory were presented in Section 4.2.4.1. Using the same evaluation process yields a dose of 52 person-rem (dry storage in existing facilities). The associated worker population risk for this part of the preferred alternative is 0.021 LCF.

The incident-free radiological worker health impacts at the Savannah River Site due to the preferred alternative are assumed to be equal to those discussed in Section 4.3.6.6.4 under the subheading, *Incident-Free Impacts at the Savannah River Site*. The highest estimated worker population risk is 0.21 LCF.

The total of the onsite activities' estimated incident-free worker population risks at both sites is 0.23 LCF, which means that there would be an approximately 23 percent chance of one additional LCF among the affected radiation workers at the two sites.

Impacts to the Public of Accidents Onsite

Accident scenarios, frequencies, consequences, and risks for the preferred alternative at the Idaho National Engineering Laboratory are the same as those for the basic implementation of Management Alternative 1. The estimated accident frequencies and consequences are presented in Table 4-20. The highest estimated public MEI/NPAI consequence is 0.000015 LCF and the highest estimated public population consequence is 1.0 LCF. Annual accident risks for receipt, unloading, and storage at the Idaho National Engineering Laboratory are presented in Table 4-25. Multiplying these figures by the appropriate duration of the activity (13 years for receipt and 40 years for storage) yields the risk due to accidents at the Idaho National Engineering Laboratory. These estimates are conservative because the TRIGA spent nuclear fuel involved would release fewer fission products than would the bounding radionuclide inventory presented in Appendix B, Table B-6 that was used for the evaluations in the basic implementation of Management Alternative 1. The highest estimated accident MEI/NPAI risk for this part of the preferred alternative is 0.0000019 LCF, which is due to a criticality event at an existing wet storage facility. The highest estimated accident population risk for this part of the preferred alternative is 0.016 LCF, which is due to an accidental fuel assembly breach.

The radiological public health impacts due to accidents at the Savannah River Site under the preferred alternative are assumed to be equal to those discussed in Section 4.3.6.6.4 under the subheading, *Impacts of Chemical Separations Accidents at the Savannah River Site*. The highest estimated public MEI risk is 0.000047 LCF and the highest estimated public population risk is 0.43 LCF.

The maximum of the onsite activities' estimated accident public MEI risks is equal to 0.000047 LCF, which would occur at the Savannah River Site. The chance of this hypothetical individual incurring an LCF due to the preferred alternative would be less than one in ten thousand.

The total of the onsite activities' estimated accident population risks at both sites is equal to 0.45 LCF. This means that there would be an approximately 45 percent chance that one additional LCF would be incurred among the people living near both sites due to accidents during these activities.

4.7.5.2 Waste Management

Implementation of the receipt and storage portions of the preferred alternative would introduce a very small increase in waste generation over current levels at both sites. Baseline site generation of waste is shown in Appendix F, Tables F-23 and F-46 for the Savannah River Site and the Idaho National Engineering Laboratory, respectively. It should be noted that the figures represent storage of more fuel elements, at both sites, than the amounts indicated by the preferred alternative. Implementation of a new technology would produce waste in the amounts presented in Table 4-56.

If the chemical separation portion of the preferred alternative is implemented, this would generate different wastes at the Savannah River Site in place of some of the waste from the new technology. As discussed in Section 4.3.6.6.5, the primary wastes generated during conventional chemical separation and vitrification operations are high-level waste glass in canisters and saltstone. Assuming the chemical separation portion of the preferred alternative could involve up to approximately one-third of the aluminum-based foreign research reactor spent nuclear fuel (6,000 elements), this waste generation would be about one-third of the amount generated under Implementation Alternative 6. Under the preferred alternative, DOE could generate up to approximately 24 high-level waste glass canisters and 1,350 cubic meters (47,700 cubic feet) of saltstone. These wastes would be managed along with much larger quantities of identical wastes in existing facilities at the Savannah River Site.

4.7.5.3 Cumulative Impacts

Cumulative impacts from the implementation of the preferred alternative at both the Idaho National Engineering Laboratory and the Savannah River Site are expected to be lower than those presented for the basic implementation of Management Alternative 1 in Sections 4.2.4.3.1 and 4.2.4.3.2 for the two sites, respectively. At both sites the cumulative impacts from the management of foreign research reactor spent nuclear fuel and impacts from other existing or planned activities or actions at the sites, as presented in Tables 4-29 and 4-30 for Savannah River Site and Idaho National Engineering Laboratory, respectively, including activities not related to the management of spent nuclear fuel, would not challenge or have detrimental effects on the public or environmental resources at the sites.

4.7.5.4 Mitigation Measures

Although environmental impacts at both the Savannah River Site and the Idaho National Engineering Laboratory for the implementation of the preferred alternative would be minimal in all environmental media and mitigation measures would not be necessary, the sites would implement measures in some areas to minimize impacts. Such measures would be taken in the areas of pollution control, socioeconomics, cultural resources, air and water resources, occupational and public health and safety, and accident prevention. Section 4.2.4.6 provides details on these issues.

4.7.5.5 Environmental Justice

The environmental justice conclusions for the management sites discussed in Section 4.2.4.5 for the implementation of Management Alternative 1 are valid for the preferred alternative. As discussed in Section 4.2.4.5, minority or low-income populations living near the Savannah River Site or the Idaho National Engineering Laboratory would not be subjected to any disproportionately high and adverse impacts.

4.7.6 Short Term Uses and Long Term Productivity

The use of land at the Savannah River Site for the potential construction of the new technology facilities would conform with the land use policy at the site. After adoption of an overall strategy for the management of all DOE-owned spent nuclear fuel (including spent nuclear fuel from foreign research reactors), some of the areas may be released for other productive uses.

4.7.7 Irreversible and Irretrievable Commitments of Resources

The operation of existing storage facilities at both sites would involve the consumption of some irretrievable amounts of electrical energy. The potential construction of new technology facilities at the Savannah River Site would consume irretrievable amounts of electrical energy, fuel, concrete, sand, and gravel. Other resources used in the 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.

4.7.8 Summary of the Impacts of the Preferred Alternative

The principal impacts of the preferred alternative would be occupational and public health and safety impacts. These are presented in Table 4-63 in terms of the risk of death due to cancer during each of the four segments of this alternative. The table also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF due to the preferred alternative. The population risk expresses the estimated number of additional LCF among the entire potentially exposed population.

Table 4-63 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the management site receiving the maximum dose allowed by DOE regulation every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free risk for individual members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 0.0000043 LCF.

The highest estimated accident MEI risk is 0.000047 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to an accident under this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-63, the total incident-free population risk would be 0.25 LCF for the potentially exposed public, while the corresponding risk would be 0.30 LCF for workers. Thus, there would be an estimated 25 percent chance of incurring one additional LCF among the exposed general public, and a 30 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Table 4-63 Maximum Estimated Radiological Health Impacts of the Preferred Alternative

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 7.1×10^{-7}	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2.9×10^{-10}	7.1×10^{-7}	---
Ground Transport			
Incident-Free	0.00052	0.067	0.022
Accidents	2.7×10^{-11}	0.00072	---
Site Activities			
Incident-Free	0.026	0.18	0.23
Accidents	0.000047	0.45	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000047	----	----
Total Population Risk			
Incident-Free	----	0.25	0.30
Accidents	----	0.45	----

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-63. There is approximately a five percent chance that a truck driver or member of the public could die in a traffic accident associated with the preferred alternative. This death would be unrelated to the radioactive nature of the cargo.

4.8 Comparison of the Alternatives

This chapter has identified the policy considerations and potential environmental impacts resulting from the proposed action, with all of its various alternatives, and the No Action Alternative. This section provides a comparison of the potential impacts of each alternative, with emphasis on key issues such as the amount of HEU removed from international commerce and risks to workers and the public.

4.8.1 Amount of HEU Removed from International Commerce

The purpose and need for Agency action is driven by the concern that HEU in civilian commerce might be diverted into a nuclear weapons program. Removal of HEU from international civilian commerce will greatly enhance the goals of the U.S. nuclear weapons nonproliferation policy. Figure 4-21 compares the quantities of HEU that would be removed from international civil commerce under the basic implementation of Management Alternative 1, the implementation alternatives, the Hybrid Alternative, the No Action Alternative, and the preferred alternative.

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would remove up to an estimated 4.6 metric tons (5.1 tons) of HEU from international commerce. By accepting this weapons-grade material into the United States for storage, the risk of material diversion would be eliminated. For comparison, the United States moved about 0.6 metric tons (0.7 tons) of HEU from Kazakhstan to the United States in November and December 1994 to ensure that it could not be diverted into a nuclear weapons program. The quantity of HEU involved in the basic implementation of Management Alternative 1 is over seven times the amount removed from Kazakhstan. The HEU in foreign research reactor spent nuclear fuel, however, is mixed with fission products, so it would require more sophisticated chemical processing to convert it to uranium metal suitable for use in nuclear weapons.

Implementation Alternatives: Acceptance of amounts of foreign research reactor spent nuclear fuel different from the amounts identified in the basic implementation of Management Alternative 1 could have an impact on the amount of HEU in international civil commerce. As shown in Figure 4-21, the implementation alternative of accepting spent nuclear fuel only from developing nations would remove up to only about 0.24 metric tons (0.26 tons) of HEU from international commerce. The implementation alternative of accepting target material in addition to the foreign research reactor spent nuclear fuel in the basic implementation of Management Alternative 1 would remove the most HEU (up to 4.8 metric tons or 5.3 tons) from international commerce. If the acceptance policy lasted for only 5 years, then the amount of HEU involved would be only up to 4.1 metric tons (4.5 tons).

Implementation through financial arrangements different from those identified in the basic implementation of Management Alternative 1 could indirectly impact the amount of HEU removed from international commerce depending on whether those financial adjustments influence the amount of foreign research

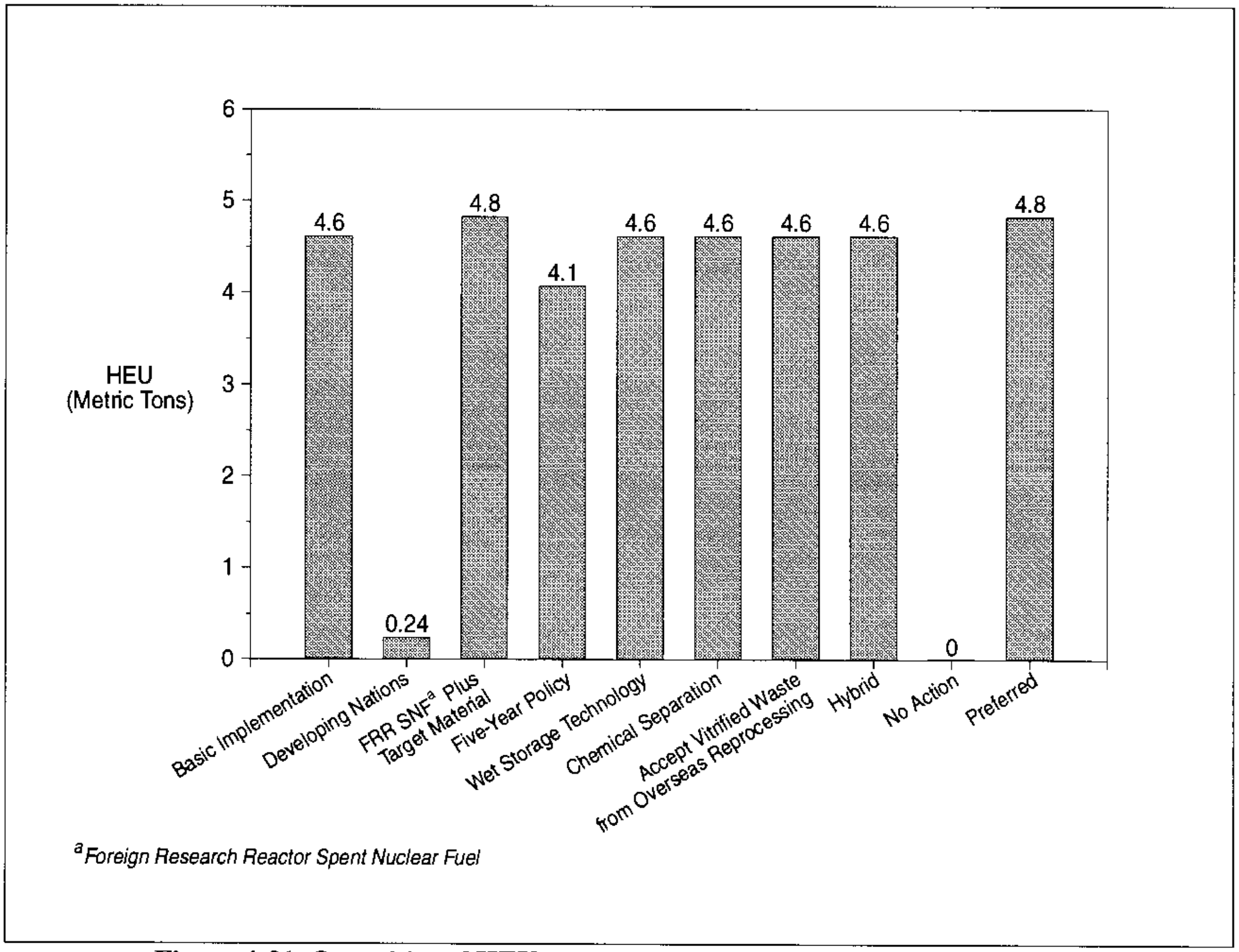


Figure 4-21 Quantities of HEU that Would Be Removed from International Commerce Under Each Alternative

reactor spent nuclear fuel transported to the United States. The final amount of HEU removed from international civil commerce through the application of different financial arrangements cannot be readily determined at this point.

Implementation by taking title to the foreign research reactor spent nuclear fuel at locations different from those identified in the basic implementation of Management Alternative 1 would not change the amount of HEU removed from international commerce, i.e., the action would still remove up to 4.6 metric tons (5.1 tons) of HEU. Similarly, the use of wet storage technology for the interim period instead of dry storage technology as identified in the basic implementation of Management Alternative 1 would not change the amount of HEU removed from international civil commerce, since the alternative relates to actions within the United States. Implementation by use of near term chemical separation in the United States instead of interim storage would also cause no change in the amount of HEU removed, again because the alternative involves actions in the United States.

Storing foreign research reactor spent nuclear fuel at one or more overseas sites would have a questionable effect on the amount of HEU removed from international commerce. Although this management alternative would provide the United States some limited measure of control over the foreign research reactor spent nuclear fuel, the prevention of material diversion into a nuclear weapons program would not be as fully ensured as if the foreign research reactor spent nuclear fuel was accepted into the United States. This alternative would leave HEU stockpiled around the world.

The implementation alternative of overseas reprocessing would remove the same amount of HEU from international commerce as would the basic implementation of Management Alternative 1, independent of decisions on the management of the resulting high-level waste.

Hybrid Alternative: The Hybrid Alternative chosen for analysis would remove the same amount of HEU from international commerce as would the basic implementation of Management Alternative 1, independent of decisions on the management of the resulting high-level waste.

No Action Alternative: Under this alternative, the United States would rely solely on the foreign governments' compliance with international agreements to control the foreign research reactor spent nuclear fuel. A policy of no action by DOE and the Department of State runs counter to U.S. nuclear weapons nonproliferation policy by causing continued reliance on HEU, thus not realizing the goal of eliminating civil commerce in HEU.

Preferred Alternative: The preferred alternative would remove the same amount of HEU (up to 4.8 metric tons or 5.3 tons) from international commerce as would Implementation Alternative 1c of Management Alternative 1. This amount is higher than for the other alternatives.

4.8.2 Radiological Risk to Individuals

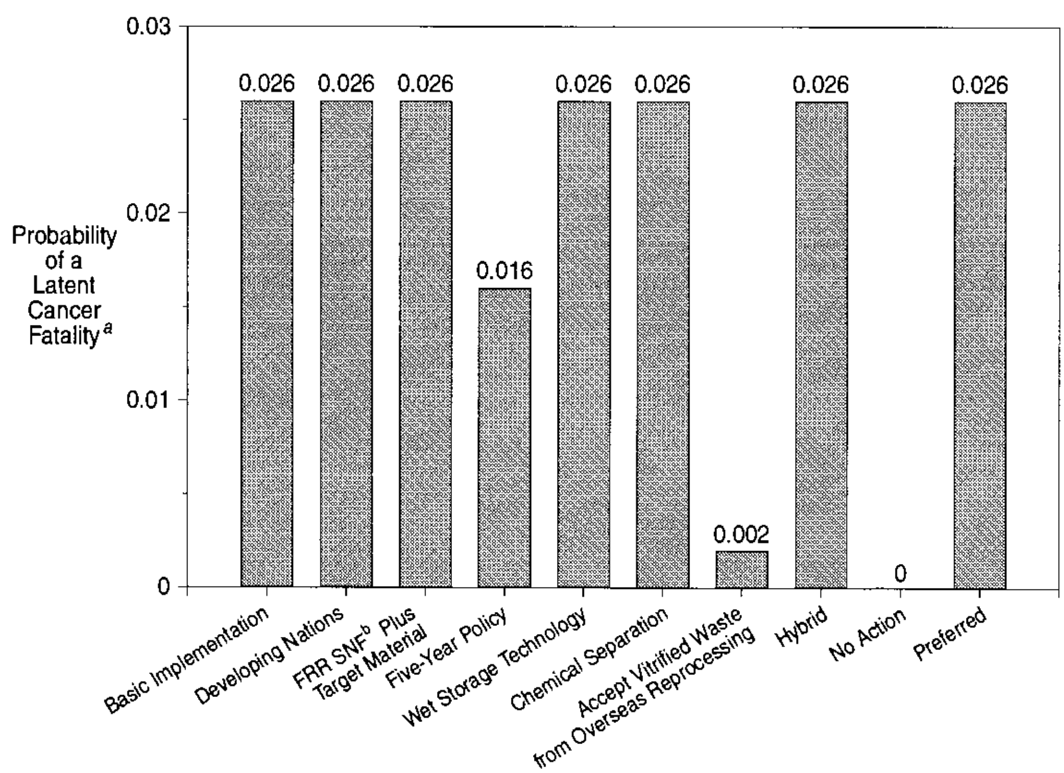
A maximally exposed worker or an MEI in the public is a hypothetical individual who records the highest possible exposure to radiation in a given situation, and the associated risks are different depending on the alternative considered. Figures 4-22 and 4-23 present comparisons of the estimated radiological risk to the maximally exposed worker and to the MEI under each alternative for incident-free and accident conditions, respectively. Alternatives involving the smallest number of cask shipments into the United States would produce the lowest individual risks. There would be no maximally exposed worker risk or MEI risk in the United States under the No Action Alternative.

The incident-free maximally exposed worker risk estimates are driven by the assumption that a radiation worker would receive the maximum radiation dose allowed by law for every year that foreign research reactor spent nuclear fuel is accepted. This risk depends only on the duration of the action, not on the number of casks or elements. Thus, the Five-Year Acceptance Alternative would present lower risk than the alternatives which last for 13 years.

The accident MEI risk estimates are dominated by onsite accident scenarios. This is because during marine transport, port activities, and ground transport, the foreign research reactor spent nuclear fuel would be inside transportation casks. During onsite activities, while spent nuclear fuel is outside of transportation casks, the probability of an incident that could release radioactive material is higher. The highest estimated accident MEI risk in the public is 0.00015 LCF, which means that this hypothetical individual's increased chance of incurring an LCF would be less than two in ten thousand.

4.8.3 Radiological Risk to Exposed Populations

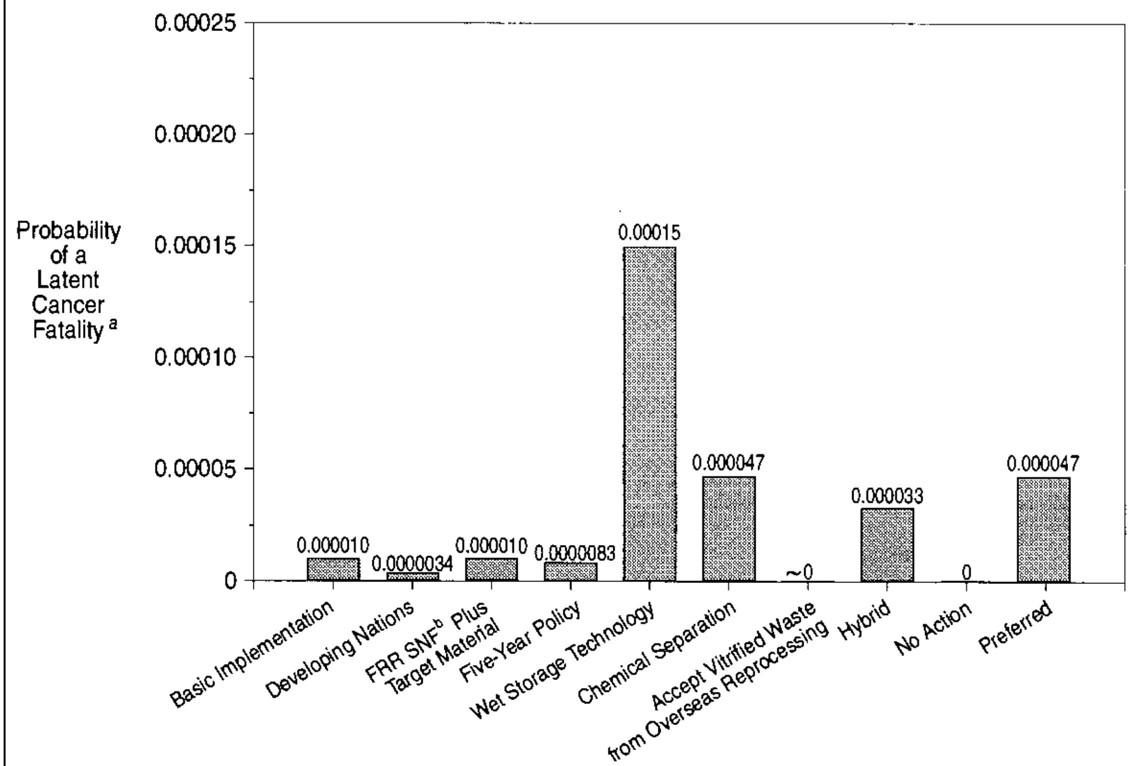
Population risk is the risk of additional latent cancers occurring among people (both public and workers) who would be exposed to radiation. Risks vary with the alternative considered. Figures 4-24 and 4-25 present comparisons of the estimated incident-free radiological risks to the public and worker populations under each alternative. Alternatives involving the smallest number of cask shipments into the United



[a] *Impacts evaluated were those in the United States and on the Global Commons.*

[b] *Foreign Research Reactor Spent Nuclear Fuel*

Figure 4-22 Maximum Estimated Incident-Free Radiological Risk to the Maximally Exposed Worker Under Each Alternative



[a] Impacts evaluated were those in the United States and on the Global Commons.

[b] Foreign Research Reactor Spent Nuclear Fuel

Figure 4-23 Maximum Estimated Accident Radiological Risk to the MEI in the Public Under Each Alternative

States would produce the lowest population risks. The chemical separation, overseas reprocessing, and preferred alternative are the alternatives in which the waste would be conditioned for disposal. Under the other alternatives, some form of processing may be required at some time in the future before disposal. There would be no population risk in the United States under the No Action Alternative. Under all the alternatives the estimated incident-free public and worker population risks would result in less than one-half additional LCF among each population group.

Figure 4-26 presents a comparison of the estimated accident radiological population risks to the public under each alternative. Those alternatives involving some form of processing in the United States would present the largest accident risks, but these risks would occur in the near term. Under the other alternatives, some form of processing may be required at some time in the future before disposal. Under all the alternatives, the estimated accident public population risks would result in less than one-half additional LCF.

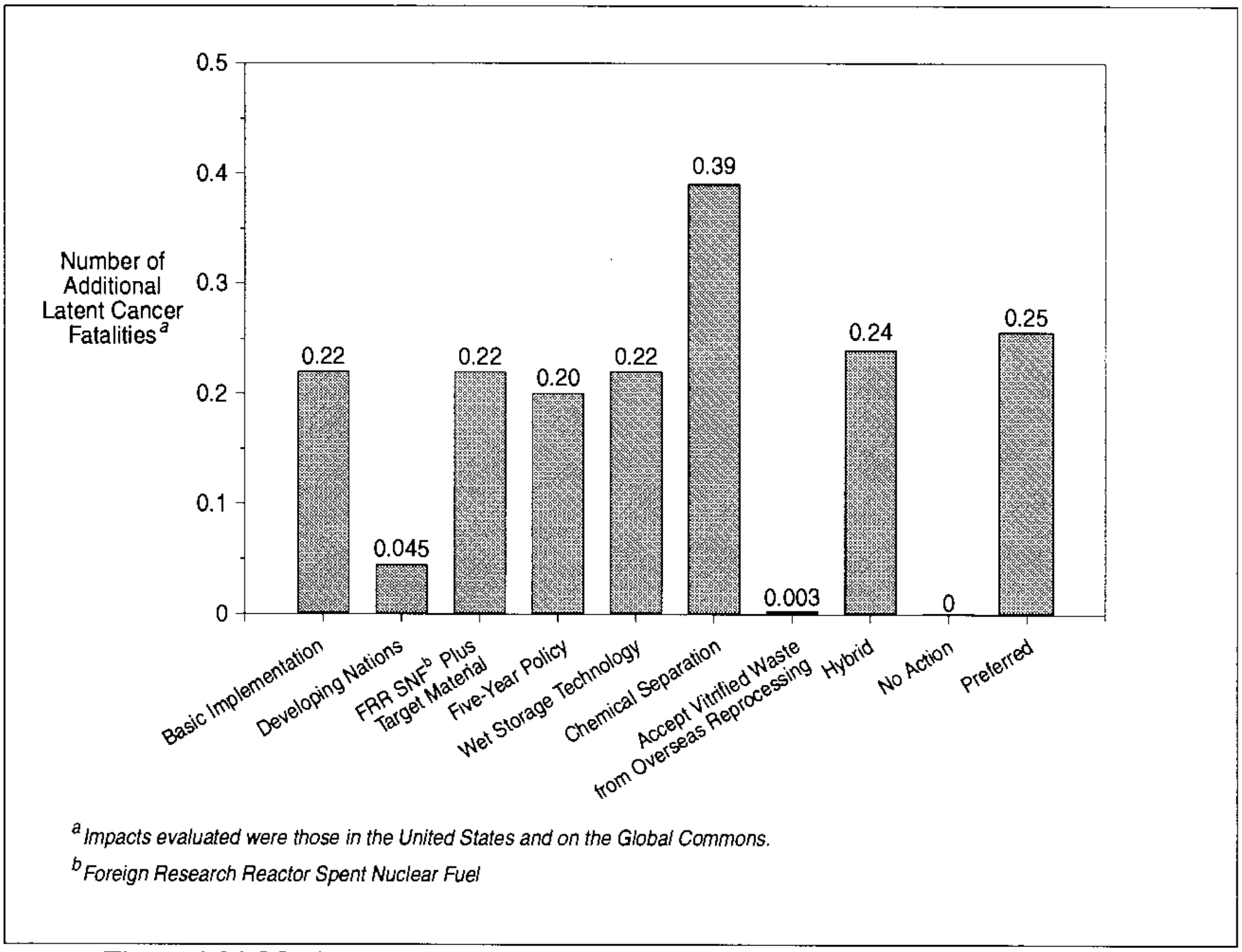


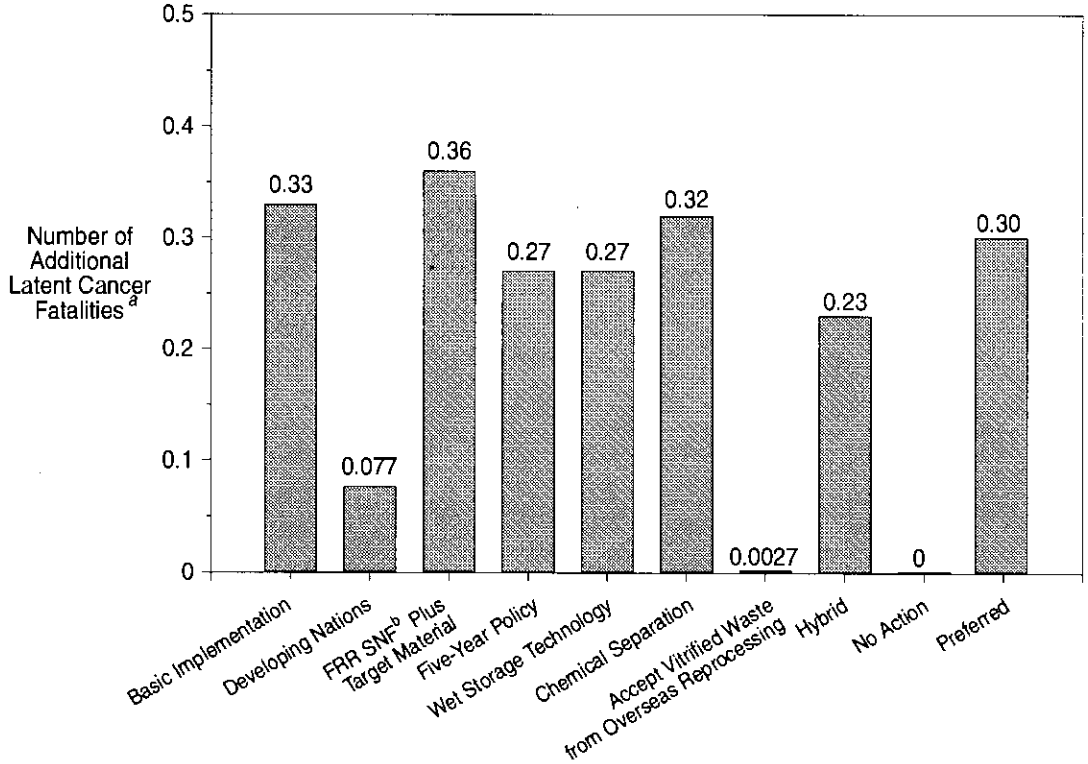
Figure 4-24 Maximum Estimated Incident-Free Radiological Population Risk to the General Public Under Each Alternative

4.8.4 Nonradiological Risks

The transport of foreign research reactor spent nuclear fuel from the ports to the sites would involve some risk of death due to traffic accidents for both the truck drivers and the public. Figure 4-27 presents a comparison of the estimated traffic accident risk to both the drivers and public combined under each alternative. Estimates include the risks associated with transporting the empty casks back to the ports.

Results are directly proportional to the number of highway miles over which casks would be transported under each alternative. The basic implementation of Management Alternative 1 and four of the implementation alternatives would have essentially the same risk, while the Developing Nations Subalternative and the Hybrid Alternative would have lower traffic accident risks.

Under the subalternative of accepting vitrified waste from overseas reprocessing, an estimated eight cask shipments would be accepted in the United States, so the traffic accident risk would be extremely low. There would be no population risk in the United States under the other overseas subalternative, as well as the No Action Alternative.



[a] *Impacts evaluated were those in the United States and on the Global Commons.*

[b] *Foreign Research Reactor Spent Nuclear Fuel*

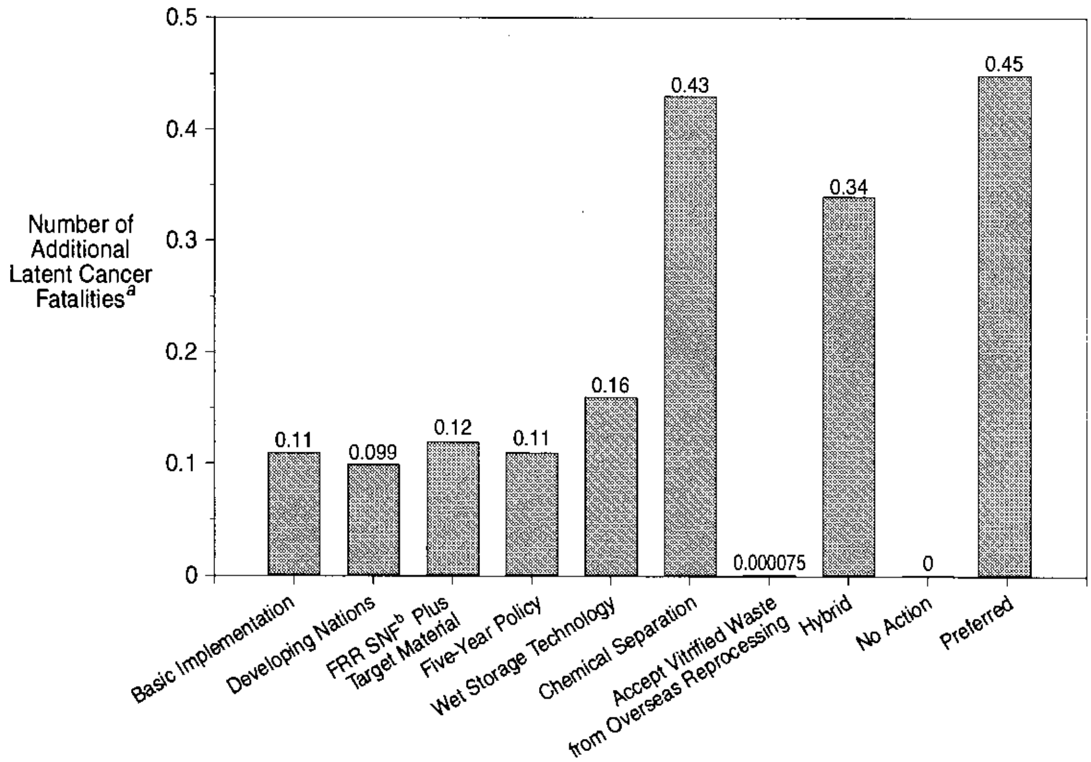
Figure 4-25 Maximum Estimated Incident-Free Radiological Population Risk to Workers Under Each Alternative

The traffic accident risk is also relatively low under the preferred alternative because all the cask shipments of aluminum-based foreign research reactor spent nuclear fuel would go through an east coast port or ports to the Savannah River Site. This effectively minimizes the ground transport risk by minimizing the number of highway miles required.

4.8.5 Land Use

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would not result in major land use issues at any of the potential foreign research reactor spent nuclear fuel management sites. If additional storage space were required for the foreign research reactor spent nuclear fuel, the space would be built on DOE-owned lands, inside the boundaries of the DOE management sites.

Implementation Alternatives: Acceptance of amounts of foreign research reactor spent nuclear fuel different from the amount identified in the basic implementation of Management Alternative 1 would not cause land use issues, even though storage needs may vary due to the United States receiving a larger (if target material is accepted in addition to spent nuclear fuel) or smaller (e.g., from developing nations only)



[a] *Impacts evaluated were those in the United States and on the Global Commons.*

[b] *Foreign Research Reactor Spent Nuclear Fuel*

Figure 4-26 Maximum Estimated Accident Radiological Population Risk to the General Public Under Each Alternative

amount of material than identified in the basic implementation of Management Alternative 1. As mentioned above, additional storage space, if required, would be created on DOE-owned land, creating no outside land use issues.

Acceptance of foreign research reactor spent nuclear fuel for periods of time different from the time periods identified in the basic implementation of Management Alternative 1 would not cause any land use issues as the timeframe would not necessarily change the amount of foreign research reactor spent nuclear fuel received by the United States. If a policy of 5 years of acceptance was instituted, less spent nuclear fuel would be received by the United States, and if an indefinite HEU/10-year LEU policy were to be adopted, storage space would be created on DOE management sites, causing no issues in relation to outside lands.

Implementation through financial arrangements different from those identified in the basic implementation of Management Alternative 1 would have no impact on land use, as this alternative would have no effect on lands not owned by DOE.

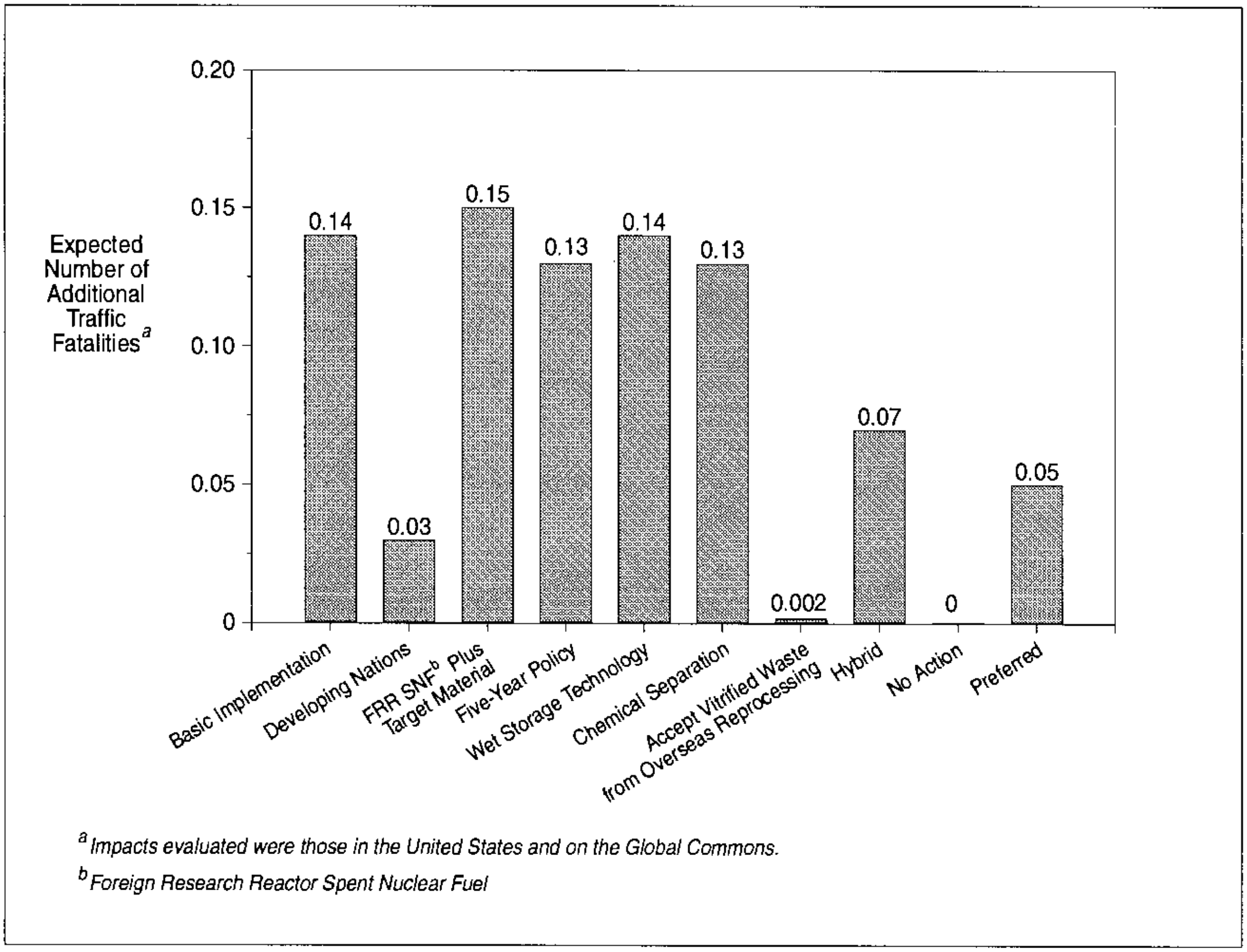


Figure 4-27 Maximum Estimated Traffic Accident Risk Under Each Alternative

Implementation by taking title to the foreign research reactor spent nuclear fuel at locations different from those identified in the basic implementation of Management Alternative 1 would cause no land use issues, as it would have no effect on the storage needs or the amount of foreign research reactor spent nuclear fuel received by the United States.

Use of wet storage technology for the interim period instead of dry storage technology as identified in the basic implementation of Management Alternative 1 would cause no land use issues, as the storage facilities (wet or dry) would be on DOE-owned land, and would have no effect on outside (non-DOE-owned) lands. If DOE decides to purchase the BNFP facility for interim wet storage, however, this would require adding some land to the Savannah River Site.

Implementation by use of near term chemical separation in the United States instead of interim storage would have no impact on land use, as the separation would be performed on DOE-owned land, with no effect on outside (non-DOE-owned) lands.

Similarly, there would be no land use concerns under either of the overseas subalternatives or the Hybrid Alternative presented in this EIS. A policy of no action (the No Action Alternative) regarding foreign research reactor spent nuclear fuel would cause no land use issues in the United States.

Land use for construction under the preferred alternative would be similar to the land use for construction under the basic implementation of Management Alternative 1.

4.8.6 Cultural Resources

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would not result in major impact to the cultural resources of the management sites being considered for the storage of the foreign research reactor spent nuclear fuel. Although the sites have not been evaluated and audited for cultural resources, surveys would be completed prior to any construction or other activity that would potentially disturb these areas. Areas of cultural or historical significance are protected by laws and acts (e.g., Native American Grave and Repatriation Act, National Historic Preservation Act, etc.), and the basic implementation of Management Alternative 1 is not likely to have an impact on areas of cultural or historical significance.

Implementation Alternatives: Since the safety of areas of cultural or historic significance is protected under the basic implementation of Management Alternative 1, these areas would not be impacted by any of the various implementation alternatives, the Hybrid Alternative, or the preferred alternative.

The overseas subalternatives would have no impact on cultural resources, as these subalternatives involve no use of DOE management sites. Similarly, the No Action Alternative would have no impact for the same reason.

4.8.7 Air Quality

While all possible precautions and safeguards would be utilized in an effort to conserve air quality, it would be impacted by U.S. acceptance of foreign research reactor spent nuclear fuel.

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would not be expected to have major impacts on air quality, and projected emissions from foreign research reactor spent nuclear fuel storage at management sites would not violate Federal or State standards. Dust from construction activities could be controlled with standard techniques. Particulate emissions could have temporary effects on localized visibility, but would not adversely affect Federal or State attainment standards.

Implementation Alternatives: Air quality would be most affected under Implementation Alternative 6, the Hybrid Alternative, or the preferred alternative, all of which involve the use of some form of processing in the United States. Chemical separation would yield a higher effect on air quality than any of the other implementation alternatives.

Since the two overseas subalternatives deal strictly with spent nuclear fuel management overseas, and the No Action Alternative involves no action on the part of DOE or the Department of State, air quality in the United States would not be affected under these alternatives.

4.9 Costs

The costs of implementing various scenarios of the proposed action, including the preferred alternative, plus disposal are presented in this section. Additional details pertaining to costs are provided in Appendix F, Section F.7. For the purpose of the cost analysis, the alternatives described in Section 2.1 were adjusted to reflect the Record of Decision on the Programmatic SNF&INEL Final EIS (DOE, 1995c) issued in May 1995. According to this Record of Decision, if foreign research reactor spent nuclear fuel is managed in the United States, the aluminum-based portion would be managed at the Savannah River Site and the TRIGA portion would be managed at the Idaho National Engineering Laboratory. DOE selected six scenarios, including the preferred alternative, for cost analysis. The costs of disposal were estimated for each scenario and are included in the analysis. The cost analysis also considers the financing arrangements discussed in Sections 2.2.1.2 and 2.2.2.3 that would affect the cost to the United States.

All costs are presented in two parts: 1) minimum discounted costs (base case) for the well-defined program components and integration approaches, and 2) "other cost factors" that are likely but sufficiently uncertain that they cannot be directly included in the minimum discounted costs. For the preferred alternative, however, a wide range of costs is presented because of the uncertainty associated with the new technology development program. An example of an item covered by "other cost factors" would be the cost growth caused by adverse weather that extends the time required to make shipments of the foreign research reactor spent nuclear fuel. The costs are shown as net present values in a consistent accounting framework.

4.9.1 Scenarios Analyzed

For the purpose of the cost analysis, six scenarios were analyzed. The scenarios reflect the alternatives that affect cost directly, and are consistent with the Record of Decision of the Programmatic SNF&INEL Final EIS (DOE, 1995c). The six cost scenarios are:

1. *Management Alternative 1 (Storage)* — Storage of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site in new dry or wet storage facilities; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory in existing wet or dry storage facilities.

2. *Management Alternative 1 (revised to incorporate chemical separation)* — Chemical separation of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

3. *Target Material* — Storage of target material at the Savannah River Site. This scenario provides the cost differential that can be used to assess the cost of managing target material in addition to the foreign research reactor spent nuclear fuel in Management Alternative 1 storage and chemical separation scenarios.

4. *Management Alternative 2* — Management of all foreign research reactor spent nuclear fuel overseas. This scenario reflects a combination of reprocessing and dry storage overseas. Countries with the capability to accept the waste from reprocessing are assumed to have their spent nuclear fuel reprocessed. The rest use dry storage.

5. *Management Alternative 3* — Chemical separation of a portion of the aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; reprocessing of the remainder of aluminum-based foreign research reactor spent nuclear fuel overseas; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

6. *Preferred Alternative* - Implementation of a new treatment and/or packaging technology for aluminum-based foreign research reactor spent nuclear fuel and target material at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

By varying the quantities of material managed in different ways in the United States and overseas, different cost scenarios can be generated. The costs of these variations are bounded by the costs of the scenarios described above. For instance, a management alternative that includes acceptance of target material into the United States would be represented by a combination of Scenarios 1 and 3 or 2 and 3.

The implementation alternatives under Management Alternative 1 related to alternative amounts of foreign research reactor spent nuclear fuel eligible under the policy (Section 2.2.2.1), and alternative policy durations (Section 2.2.2.2), were not considered separately in the cost analysis because they are bounded by the cost scenarios analyzed. These implementation alternatives reduce the amount of foreign research reactor spent nuclear fuel eligible under the policy.

The implementation alternative under Management Alternative 1 related to alternative locations for taking title to the foreign research reactor spent nuclear fuel (Section 2.2.2.4) was not considered because it does not affect the cost analysis.

4.9.2 Minimum Program Costs

Table 4-64 shows the minimum discounted program costs (base case) for the six scenarios defined above. These costs cover all foreign research reactor spent nuclear fuel shipments, management over 40 years, and geologic disposal. Uncertainties (risks) and escalation are zero. Costs to manage target material (Scenario 3) could be added to the costs of Scenarios 1, 2, 4, and 5 to produce a minimum program cost. Costs to manage target material are included in the preferred alternative (Scenario 6).

Table 4-64 Minimum Program Costs (Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Net Present Value*
1. Management Alternative 1 (Storage)	725/775[a]
2. Management Alternative 1 (revised to incorporate Chemical Separation)	625
3. Target Material	35
4. Management Alternative 2	1,250
5. Management Alternative 3	675
6. Preferred Alternative[b]	625-950

[a] *Dry/Wet new storage facilities*

[b] *Includes target material*

The schedule for activities in Europe under Scenario 5 is similar to that in the United States but not exactly the same. Reprocessing takes place over 13 years at Dounreay (the same timespan used for chemical separation at the Savannah River Site) although it could be completed at Dounreay in 9 or 10 years. Dounreay's charges for reprocessing are based on 1996 costs, not costs for 1996 through 2008 averaged over the 13 year period (as was done for the Savannah River Site). Geologic disposal takes place in 2025 through 2030 in Europe and 2030 through 2035 in the United States.

Costs are discounted at 3 percent for the portion to be managed overseas and at 4.9 percent for the portion to be managed in the United States. These net present values imply that all funds required to pay for the program over its 40-year life are received and placed in a trust fund accruing interest at a 4.9 percent real rate of return. This rate of return is required by the Office of Management and Budget for the year ending February, 1996.

Because of the uncertainties involved with the implementation of the new technology, the cost for Scenario 6 (preferred alternative) is presented as a range as discussed in Appendix F, Section F.7.2.9. Also, the shipping costs in Scenario 6 include the assumption that only 38 cask shipments would be accepted on the West Coast.

4.9.3 Other Cost Factors

There are four important sources of cost risk (excluding escalation) that are not part of the minimum costs in Table 4-64. Table 4-65 shows the likely values (risks) for these factors, taking into account the absolute values of the uncertainties and their probability of occurrence.

Table 4-65 Other Cost Factors (Net Present Value, Millions of 1996 Dollars in 1996)

	Cost Factors				
Scenario	*Systems Integration & Logistics*	*Component Risks*	*Non-program Risks*	*3% Discount Rate*	*Range*
1. Management Alternative 1 (Storage)	100	75	35	175	385
2. Management Alternative 1 (revised to incorporate Chemical Separation)	100	±15	10	125	200-250
3. Target Material	5	5	0	25	35
4. Management Alternative 2	100	±500	1000	250	350-1850
5. Management Alternative 3	100	±10	150	75	315-335
6. Preferred Alternative[b,c]	100	75	35	225	435

[a] *It is assumed that risks are the same for dry or wet storage options.*

[b] *Includes target material*

[c] *It is assumed that risk factors are the same as Management Alternative 1 (Storage)*

The other cost factors summarized in Table 4-65 are as follows:

1. *Systems Integration and Logistics Risks* - Significant risks exist in the details of the policy implementation. The implementation of the policy would involve up to 41 foreign countries, 13 years of possible receipts, dozens of foreign ports, up to ten domestic ports, two U.S. management sites, and possibly several new facilities. Technical and procedural bottlenecks could arise in several areas.

2. *Component Risks* - Significant risks exist for specific components of the foreign research reactor spent nuclear fuel program, e.g., the comprehensiveness of the acceptance criteria for aluminum-clad spent nuclear fuel characterization for dry storage, the methods of spent nuclear fuel disposal, the cost allocation at existing and new facilities, and development of new technology.

3. *Non-Program Risks* - Significant risks exist for components of other programs that affect the implementation of the foreign research reactor spent nuclear fuel EIS, (e.g., escalating repository costs, adoption of monitored retrievable storage, and differences in facility utilization plans between this EIS and those of other EISs affecting the Savannah River Site and the Idaho National Engineering Laboratory). For Scenario 5, the risks are that no spent nuclear fuel infrastructure exists in more than half of the eligible countries and that no geologic disposal program exists in most of the eligible countries.

4. *Discount Rate Risks* - Significant risks exist that the current discount rate required by the Office of Management and Budget for the year ending February, 1996 (4.9 percent real) will be reduced to a more historically representative level (e.g., 3 percent) in some future annual update. The base case costs for management outside the United States are discounted at a 3 percent rate. The use of a high discount rate is particularly risky because 1) revenues are likely to be fixed (in $/kgTM) early in the program while expenses are variable and uncertain, and 2) revenues received from the reactor operators during the 1996 through 2008 shipping period will almost certainly exceed the costs of management activities during that period. Mathematically, the excess revenues are placed in a trust fund that compounds interest at the discount rate. If the discount rate exceeds the rate at which funds are actually likely to compound, then outyear program costs (e.g., disposal) could not be met from the principal and accrued interest in the trust fund. A reduction in the discount rate from 4.9 percent to 3.0 percent has a larger impact on the program than any of the technical or systems integration risks.

4.9.4 Potential Total Costs

Table 4-66 combines the base case costs with the "other cost factors" to provide a realistic expectation of the potential total costs of the program, excluding escalation. The "other cost factors" are divided into technical factors and discount rate-related factors. This table also shows the cumulative percentage effect on the minimum discounted program costs of real escalation at a rate of 1 percent per year over 40 years.

Table 4-66 Potential Total Costs (Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Minimum Program Cost*	*Other Cost Factors (Technical)*	*Other Cost Factors (Discount Rate)*	*Potential Total Cost, No Escalation*	*1% Real Escalation, Cumulative*
1. Management Alternative 1 (Storage)	725/775[a],	210	175	~1,100	+11%
2. Management Alternative 1 (revised to incorporate Chemical Separation)	625	85-145	125	~900	+9%
4. Management Alternative 2	1,250	600-1,600	250	2,100-3,100	+13%
5. Management Alternative 3[c]	675	225-275	75	~1000	+9%
6. Preferred Alternative[b]	625-950	210	225	~1,050-1,400	+10%-11%

[a] *Dry/Wet new storage facilities*

[b] *Includes target material*

[c] *The total cost risk to the United States is less than 1/2 the total cost risk because a large portion of the*

Table 4-66 shows that the net present value of the potential total costs of implementing the program completely in the United States, including an estimate of program risks but excluding escalation, range from about $900 million for Scenario 2, to $1.4 billion for Scenario 6. Scenario 5 has similar total program costs as Scenario 2 but higher risks for geologic disposal.

In Scenario 4, costs for storing foreign research reactor spent nuclear fuel overseas are highly speculative. In addition, the overseas storage costs are always higher than the more centralized management alternatives because of the extremely high cost of safely and securely managing and disposing of small quantities of spent nuclear fuel in dozens of countries.

The program costs presented in Tables 4-64, 4-65, and 4-66 are in constant 1996 dollars, discounted to 1996. This implies that funds required to cover these costs are received in 1996 and explicitly or implicitly placed in a trust fund. If payments into the trust fund are deferred, then they must be larger than if they had been received on January 1, 1996. For example, if payments are made in 13 equal annual installments every December 31 over the 1996 through 2008 shipping and receiving period, then the constant-dollar payments must increase by 37 percent. A composite of payment schedules, e.g., 13 years for high-income-economy country reactor operators and pay-as-you-go (for the United States) for all other costs, including other-than-high-income-economy country costs, has the effect of increasing the required constant-dollar payments by as much as 25 to 50 percent.

4.9.5 Cost to the United States

The cost of the proposed policy to the United States depends directly on the type of financing arrangement that DOE would adopt in implementing the policy and the discount rate at which revenues from reactor operators accrue interest. Alternative financing arrangements are discussed in Sections 2.2.1.2 and 2.2.2.3 of the EIS. Briefly, the financing arrangements considered are:

1. United States bears the full cost of the program for countries with other-than-high-income economies and charges a *competitive* fee to high-income-economy countries. This is the financial arrangement in the preferred alternative.

2. United States bears the full cost for all countries (*no* fee).

3. United States charges a *full-cost-recovery* fee to all countries.

4. United States bears the full cost of the program for countries with other-than-high-income economies and charges a *full-cost-recovery* fee to high-income-economy countries.

From a practical standpoint, the U.S. cost under financing arrangement 3 above would be zero. The issue would be whether any foreign countries would participate in the program if full-cost recovery exceeded a competitive fee. The first and fourth arrangements are functionally similar, the U.S. cost resulting from the difference in the *competitive* versus the *full-cost-recovery* fee. The U.S. cost under the second arrangement (*no* fee) would be the total program cost as discussed earlier. Any fees established by the United States will take place pursuant to a Federal Register notice after the Record of Decision for this EIS.

Table 4-67 shows costs to the United States for the minimum program in each of the cost scenarios analyzed (except Scenario 3) under a variety of fee schedules. Adding target material to Scenarios 1, 2, 4 or 5 would increase its costs by 3 to 4 percent. Fees of $2,000/kgTM, $5,000/kgTM, $7,500/kgTM, and $10,000/kgTM, including a pass-through of shipping charges (all levelized over 13 years), are used to provide a range of estimates for the cost to the United States. These fees do not imply that reactor operators would pay them for management in Europe or the United States, or that the fee established by the United States will be one of these values. They are used for illustration only and suggest a bounding range, exclusive of technical risk factors, discount rate adjustments, and escalation.

The cost to the United States is the sum of: 1) the cost of managing the foreign research reactor spent nuclear fuel from the other-than-high-income-economy countries, including shipping, and 2) the difference between the revenues received for management of high-income-economy country foreign research reactor spent nuclear fuel and the total program cost of managing high-income-economy country foreign research reactor spent nuclear fuel, including shipping. Including shipping in the U.S. management costs allows management costs for the United States and the United Kingdom to be presented on a comparable basis.

Table 4-67 Costs to the United States for Minimum Program Under Various Scenarios and Fee Structures (Millions of 1996 Dollars, Net Present Value of Costs in 1996, Fees Levelized over 1996-2008 Period)

Scenario[a]	Full-Cost Recovery[b]	Levelized Shipping Fee $/kgTM	Levelized Management Fee (excluding shipping) $/kgTM	Net Present Value For Levelized Fee[c] (developed countries only)				No Fee[d]	Total (excluding shipping)
				$2,000/ kgTM	$5,000/ kgTM	$7,500/ kgTM	$10,000/ kgTM	Developed Countries	
1. Management Alternative 1 (Storage)	100	1,500	6,500	325	100	(75)	(250)	475	575
2. Management Alternative 1 (revised to incorporate Chemical Separation)	90	1,500	5,800	275	50	(125)	(300)	425	525
4. Management Alternative 2[f]	500+							1,250 +	1,750+
5. Management Alternative 3[g]	85	1,500	6,000	225	75	(50)	(175)	300	375
6. Preferred Alternative[e]	90-110	1,700	5,600-9,200	275-550	50-325	(150)-125	(325)-(50)	425-700	500-800

[a] *The total mass (kgTM) of foreign research reactor spent nuclear fuel in the various scenarios is approximately as follows: Aluminum-based plus TRIGA: 115,000 kgTM; from other-than-high-income-economy countries: 15,000 kgTM; from high-income-economy countries: 100,000kgTM; to Dounreay in Scenario 5: 37,000 kgTM. The total mass of target material is approximately 3,400 kgTM, essentially all from high-income-economy countries.*

[b] *Full-cost recovery from high-income-economy countries only. The United States bears the costs of the other-than-high-income-economy countries in these cases.*

[c] *Payable in 13 equal annual installments on December 31 of the years 1996 through 2008. Add costs in column labeled "Full-Cost Recovery" to generate total cost to the United States.*

[d] *As above, implicitly paid by the taxpayers in 13 equal annual installments (to maintain consistency with the payment period of the reactor operators), excluding shipping. The net present value of shipping in Scenarios 1 and 2 is $140 Million. The net present value of shipping to the U.S. only in Scenario 5 is $90 Million. The net present value of shipping in Scenario 6 is $160 million. Adding shipping to the net present value for Scenario 2 and Scenario 5 shows that the total program costs for Scenario 5 are slightly lower.*

[e] *Includes target material*

[f] *There is no defined basis for the charges to the United States for non-U.S. management. Costs to the United States under Scenario 4 assume that the United States absorbs the cost to construct and operate independent foreign research reactor spent nuclear fuel storage installations (including all supporting safety, security, transport, health physics, etc. infrastructure) for the 22 countries with no commercial nuclear power programs and that the United States partially subsidizes the other countries, depending on their income-economy status, commercial nuclear power infrastructure, and other factors.*

[g] *Revenues paid to the United States include pass-through of shipping charges. Costs to the United States for management in Europe include the cost of blending down the HEU to LEU ($20 million).*

Table 4-67 shows that for minimum discounted program costs and fees charged to high-income-economy country reactor operators levelized over 13 years, costs to the United States for the scenarios could range from several hundred million dollars at a fee of $2,000/kgTM to a profit for fees of $7,500/kgTM to $10,000/kgTM. The cost of managing the spent nuclear fuel from the other-than-high-income-economy countries (including shipping) adds roughly $100 million more to the cost borne by the United States.

If fees in the $2,000 to $10,000 per kgTM range are established and charged over 13 years, the costs to the United States would be as estimated in Table 4-67 plus any additional cost factors not incorporated in the minimum program costs. These additional cost factors are: 1) technical risks, 2) discount rate-related risks, and 3) escalation. Table 4-66 shows that technical risks could add roughly $100 to $200 million to the costs borne by the United States. Discount rate-related risks are of a similar size. Escalation risks are uncertain but could be in the same range.

4.10 Foreign Research Reactor Spent Nuclear Fuel Risks and Common Risks

This section compares foreign research reactor spent nuclear fuel program risks to those of common activities, such as smoking, flying, receiving a medical X-ray, and so forth.

4.10.1 Risks in the Proposed Action

Preceding sections in Chapter 4 evaluated the risks from radiological and nonradiological activities and accidents in four segments: marine transport, port activities, ground transport, and site activities.

The highest estimated accident MEI risk to the general public from any of the foreign research reactor spent nuclear fuel implementation alternatives is 0.00015 LCF, as shown earlier in Figure 4-23. This would be an individual who lives at the Oak Ridge Reservation boundary under Implementation Alternative 5, Wet Storage Technology for New Construction. This hypothetical individual's chance of incurring a fatal cancer would be increased by less than two in ten thousand.

The highest estimated incident-free population risk to the general public living near any of the DOE management sites from any of the implementation alternatives is less than one-half LCF, as shown earlier in Figure 4-24. This risk occurs under Implementation Alternative 6, Near Term Chemical Separation in the United States, at the Savannah River Site. This risk would be spread among the roughly 600,000 people who live within 80 km (50 miles) of the Savannah River Site, so the average risk among these people would be less than one in a million.

The population risk to the general public due to radiation exposure during ground transport could be as high as 0.22 LCF, as discussed earlier under several of the implementation alternatives to Management Alternative 1.

Nonradiological fatalities are also unlikely. As a practical matter, the only source of nonradiological fatalities to the public is through a traffic accident with a truck or a train. Since truck or train shipments are about 100 or fewer per year, the likelihood of a crash is not high.

4.10.2 Common Radiological Risks

Table 4-68 presents several typical sources of exposure to radiation from everyday life (DOE, 1993e). The average person in the United States receives about 300 mrem each year from natural sources of radiation and about another 50 mrem from manmade sources of radiation. For example, the largest dose listed in Table 4-68 is the 200 mrem/yr from exposure to naturally-occurring radon gas. This is twice the

100 mrem/yr regulatory limit that would apply to marine workers, port workers, and truck drivers under the proposed action. It is also much higher than the dose any member of the general public would be likely to receive.

Table 4-68 Typical Sources of Radiation, Exposures, and Risks

Source	*Dose Rate (mrem/yr)*	*Risk (LCF/yr)*
Radon	200	0.0001
Internal	39	0.000020
Diagnostic X-rays	39	0.000020
Soil, rocks	28	0.000014
Cosmic rays	27	0.000014
Nuclear medicine	14	0.000007
Nuclear fuel cycle	less than 1	less than 5×10^{-7}
Fallout	less than 0.01	less than 5×10^{-9}

There are also large variations in radiation dose to which people are routinely exposed. For example, people who live at high altitudes receive more radiation dose than people who live at sea level. People who live or work in brick, granite, or marble buildings receive more radiation dose than people who live or work in wooden structures. People who live in well-insulated houses receive more radiation dose from trapped radon gas than people who live in well-ventilated houses. Taking all the various factors into account, the annual U.S. dose from background radiation can easily range from 100 mrem for people who live in well-ventilated wooden houses on sandy soil at sea level to about 1000 mrem for people who live in well-insulated houses in the Denver area (de Planque, 1994). Thus, in addition to the average annual radiation dose, routine variations in annual radiation dose are also much larger than the dose any member of the general public would be likely to receive under the proposed action.

4.10.3 Risks from Common Activities

Every activity carries some risk. Table 4-69 shows risks estimated to increase an individual's chance of death in any year by one in one million (Slovic, 1986). For example, a single airline flight across the United States would increase each passenger's radiation dose by about 4 mrem (de Planque, 1994). Most of these voluntary activities would not be considered unusually risky actions, and they can be compared to the risks presented earlier in this chapter for perspective.

Table 4-69 Risks Estimated to Increase Chance of Death in Any Year by One Chance in a Million

Activity	*Cause of Death*
Smoking 1.4 cigarettes	Cancer; heart disease
Living 2 days in New York or Boston	Air pollution
Traveling 16 km (10 mi) by bicycle	Accident
Flying 1,600 km (1,000 mi) by jet	Accident
Living 2 months in Denver on vacation from New York	Cancer caused by cosmic radiation
One chest X-ray taken in a good hospital	Cancer caused by radiation
Drinking 30 12-oz cans of diet soda	Cancer caused by saccharin

5. Applicable Laws, Regulations, and Other Requirements

5.1 Consultation

Certain Federal laws, such as the Endangered Species Act, the Fish and Wildlife Coordination Act, and the National Historic Preservation Act, require consultation and coordination by the United States Department of Energy (DOE) with other governmental entities. These consultation and coordination requirements will commence and be completed as site-specific spent nuclear fuel management projects and decisions are proposed. Any site-specific required consultations will be addressed in the site-specific Environmental Impact Statement (EIS) and/or in Volume I of DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Draft EIS (Table 5-1).

5.2 Laws and Other Requirements

This section identifies and summarizes the major laws, regulations, Executive Orders, and DOE Orders that may apply to the receipt and management of spent nuclear fuel from foreign research reactors.

Section 5.2.1 discusses the major Federal statutes that impose environmental protection and compliance requirements upon DOE. In addition, there may be other State and local measures applicable to the foreign research reactor spent nuclear fuel because Federal law delegates enforcement or implementation authority to State or local agencies. These state- and local-specific requirements are addressed in the site-specific appendices. Section 5.2.2 addresses environmentally-related Executive Orders that clarify issues of national policy and set guidelines under which Federal agencies, including DOE, must act. DOE implements its responsibilities for protection of public health, safety, and the environment through a series of Departmental Orders that are mandatory for operating contractors of DOE-owned facilities. Section 5.2.3 discusses those DOE orders related to environmental, health, and safety protection. Hazardous and radioactive materials transportation regulations are summarized in Section 5.4.2.

5.2.1 Federal Environmental Statutes and Regulations

National Environmental Policy Act (NEPA) of 1969, as amended (42 USC §4321 et seq.)

NEPA establishes a national policy promoting awareness of the environmental consequences of the activity of humans on the environment and also promoting consideration of the environmental impacts during the planning and decision making stages of a project. This Act requires all Federal agencies to prepare a detailed statement on the environmental effects of proposed major Federal actions that may significantly affect the quality of the human environment.

This EIS has been prepared in response to these NEPA requirements and policies. It discusses reasonable alternatives and their potential environmental consequences, and has been prepared in accordance with the Council on Environmental Quality and DOE regulations for implementing the procedural provisions of the NEPA Implementing Procedures (40 CFR Parts 1500 through 1508) and DOE NEPA Implementing Procedures (10 CFR Part 1021).

Table 5-1 Agency Consultations

Subject Area	*Legislation*	*Agency*
Endangered Species	Endangered Species Act of 1973, as amended; State laws	U.S. Fish and Wildlife Service, State agencies
Migratory birds	Migratory Bird Treaty Act	U.S. Fish and Wildlife Service
Bald and Golden eagles	Bald and Golden Eagle Protection Act	U.S. Fish and Wildlife Service
Archaeological, historical, and cultural preservation	National Historic Preservation Act of 1966, Archaeological Resources Protection Act, Antiquities Act, American Indian Religious Freedom Act of 1978, Native American Grave Protection and Repatriation Act of 1990	State Historic Preservation Office, President's Advisory Council, Tribes
Discharge of pollutants to water	Clean Water Act, Safe Drinking Water Act	U.S. Environmental Protection Agency, State agencies
Work in navigable U.S. waters	Clean Water Act, Rivers and Harbors Act, Coastal Management Act	U.S. Army Corps of Engineers
Prime and unique farmlands	Farmland Protection Policy Act of 1981	Soil Conservation Service
Floodplains	Executive Order 11988, Fish and Wildlife Coordination Act	U.S. Army Corps of Engineers, U.S. Fish and Wildlife Service, State agencies
Wetlands	Executive Order 11990, Fish and Wildlife Coordination Act, Clean Water Act	U.S. Army Corps of Engineers, U.S. Fish and Wildlife Service, U.S. Environmental Protection Agency, State agencies
Environmental justice	Executive Order 12898	U.S. Environmental Protection Agency
Water body alteration	Fish and Wildlife Coordination Act	U.S. Fish and Wildlife Service, State agencies
River status	Wild and Scenic Rivers Act, Anadromous Fish Conservation Act, Hanford Reach Study Act	U.S. Department of the Interior
Air pollution	Clean Air Act	U.S. Environmental Protection Agency, State and local agencies
Water use and availability	Water Resources Planning Act of 1965, Safe Drinking Water Act, and others	U.S. Environmental Protection Agency, Office of Water Policy, State agencies
Noise	Noise Pollution and Abatement Act of 1970, Noise Control Act of 1972	U.S. Environmental Protection Agency, State agencies
Siting and planning	State siting acts, county zoning regulations	State and County agencies
Waste management and transportation	Solid Waste Disposal Act, as amended by the Resource Conservation and Recovery Act and the Hazardous and Solid Waste Amendments of 1984; Comprehensive Environmental Response, Compensation, and Liability Act; Emergency Planning and Community Right to Know Act; Hazardous Materials Transportation Act	U.S. Environmental Protection Agency, U.S. Department of Transportation, U.S. Coast Guard, State agencies
Emergency Management & Response	Defense Production Act of 1950, Robert T. Stafford Disaster Relief and Emergency, Assistance Act, National Security Act of 1947	Federal Emergency Management Agency, U.S. Environmental Protection Agency, U.S. Department of Transportation, U.S. Coast Guard, State and local agencies

Atomic Energy Act of 1954, as amended (42 USC §2011 et seq.)

The Atomic Energy Act of 1954 authorizes DOE to establish standards to protect health or minimize dangers to life or property with respect to activities under its jurisdiction. Through a series of DOE Orders, DOE has established an extensive system of standards and requirements to ensure safe operation of its facilities.

Clean Air Act, as amended (42 USC §7401 et seq.)

The Clean Air Act, as amended, is intended to "protect and enhance the quality of the Nation's air resources so as to promote the public health and welfare and the productive capacity of its population." Section 118 of the Clean Air Act, as amended, requires that each Federal agency, such as DOE, with jurisdiction over any property or facility that might result in the discharge of air pollutants, comply with "all Federal, State, interstate, and local requirements" with regard to the control and abatement of air pollution.

The Act requires the Environmental Protection Agency to establish National Ambient Air Quality Standards as necessary to protect public health, with an adequate margin of safety, from any known or anticipated adverse effects of a regulated pollutant (42 USC §7409). The Act also requires establishment of national standards of performance for new or modified stationary sources of atmospheric pollutants (42 USC §7411) and requires specific emission increases to be evaluated so as to prevent a significant deterioration in air quality (42 USC §7470). Hazardous air pollutants, including radionuclides, are regulated separately (42 USC §7412). Air emissions are regulated by the Environmental Protection Agency in 40 CFR Parts 50 through 99. In particular, radionuclide emissions are regulated under the National Emission Standard for Hazardous Air Pollutants Program (see 40 CFR Part 61).

Safe Drinking Water Act, as amended [42 USC §300 (F) et seq.]

The primary objective of the Safe Drinking Water Act, as amended, is to protect the quality of the public water supplies and all sources of drinking water. The implementing regulations, administered by the Environmental Protection Agency unless delegated to the States, establish standards applicable to public water systems. They promulgate maximum contaminant levels (including those for radioactivity), in public water systems, which are defined as water systems that serve at least 15 service connections used by year-round residents or regularly serve at least 25 year-round residents. Safe Drinking Water Act requirements have been promulgated by the Environmental Protection Agency in 40 CFR Parts 100 through 149. For radioactive material, the regulations specify that the average annual concentration of manmade radionuclides in drinking water as delivered to the user by such a system shall not produce a dose equivalent to the total body or an internal organ greater than four mrem per year beta activity. Other programs established by the Safe Drinking Water Act include the Sole Source Aquifer Program, the Wellhead Protection Program, and the Underground Injection Control Program.

Clean Water Act, as amended (33 USC §1251 et seq.)

The Clean Water Act, which amended the Federal Water Pollution Control Act, was enacted to "restore and maintain the chemical, physical and biological integrity of the Nation's water." The Clean Water Act prohibits the "discharge of toxic pollutants in toxic amounts" to navigable waters of the United States. Section 313 of the Clean Water Act, as amended, requires all branches of the Federal Government engaged in any activity that might result in a discharge or runoff of pollutants to surface waters to comply with Federal, State, interstate, and local requirements.

In addition to setting water quality standards for the Nation's waterways, the Clean Water Act supplies guidelines and limitations for effluent discharges from point-source discharges and provides authority for the Environmental Protection Agency to implement the National Pollutant Discharge Elimination System permitting program. The National Pollutant Discharge Elimination System program is administered by the Water Management Division of the Environmental Protection Agency pursuant to regulations in 40 CFR Part 122 et seq. Idaho has not applied for National Pollutant Discharge Elimination System authority from the Environmental Protection Agency. Thus, all National Pollutant Discharge Elimination System permits required for the Idaho National Engineering Laboratory are obtained by DOE through Environmental Protection Agency Region 10 (40 CFR Part 122 et seq.).

Sections 401 and 405 of the Water Quality Act of 1987 added Section 402(p) to the Clean Water Act. Section 402(p) requires that the Environmental Protection Agency establish regulations for issuing permits for stormwater discharges associated with industrial activity. Although any stormwater discharge associated with industrial activity requires a National Pollutant Discharge Elimination System permit application, regulations implementing a separate stormwater permit application process have not yet been adopted by the Environmental Protection Agency.

Resource Conservation and Recovery Act, as amended (Solid Waste Disposal Act) (42 USC §6901 et seq.)

The treatment, storage, or disposal of hazardous and nonhazardous waste is regulated under the Solid Waste Disposal Act as amended by the Resource Conservation and Recovery Act and the Hazardous and Solid Waste Amendments of 1984. Pursuant to Section 3006 of the Act, any State that seeks to administer and enforce a hazardous waste program pursuant to the Resource Conservation and Recovery Act may apply for Environmental Protection Agency authorization of its program. The Environmental Protection Agency regulations implementing the Resource Conservation and Recovery Act are found in 40 CFR Parts 260 through 280. These regulations define hazardous wastes and specify hazardous waste transportation, handling, treatment, storage, and disposal requirements.

The regulations imposed on a generator or a treatment, storage, and/or disposal facility vary according to the type and quantity of material or waste generated, treated, stored, and/or disposed of. The method of treatment, storage, and/or disposal also impacts the extent and complexity of the requirements.

Current Status of Spent Nuclear Fuel under the Resource Conservation and Recovery Act

Historically, DOE chemically reprocessed spent nuclear fuel to recover valuable products and fissionable materials, and as such, the spent nuclear fuel was not a solid waste under the Resource Conservation and Recovery Act.

World events have resulted in significant changes in DOE's direction and operations. In particular, in April 1992, DOE announced the phase-out of reprocessing for the recovery of special nuclear materials. With these changes, DOE's focus on most of its spent nuclear fuel has changed from reprocessing and recovery of materials to storage and ultimate disposition. This in turn has created uncertainty regarding the regulatory status of some of DOE's spent nuclear fuel relative to the Resource Conservation and Recovery Act.

DOE has initiated discussion with the Environmental Protection Agency on the potential applicability of the Resource Conservation and Recovery Act to spent nuclear fuel. Further discussions with Environmental Protection Agency Headquarters and regional offices and State regulators are ongoing to develop a strategy for meeting any the Resource Conservation and Recovery Act requirements that might apply.

Federal Facility Compliance Act (42 USC §6921 et seq.)

The Federal Facility Compliance Act, enacted on October 6, 1992, waives sovereign immunity for fines and penalties for Resource Conservation Recovery Act violations at Federal facilities. However, a provision postpones fines and penalties after 3 years for mixed waste storage prohibition violations at

DOE sites and requires DOE to prepare plans for developing the required treatment capacity for mixed waste stored or generated at each facility. Each plan must be approved by the host State or the Environmental Protection Agency, after consultation with other affected States, and a consent order must be issued by the regulator requiring compliance with the plan. The Federal Facility Compliance Act further provides that DOE will not be subject to fines and penalties for land disposal restriction storage prohibition violations for mixed waste as long as it is in compliance with such an approved plan and consent order and meets all other applicable regulations. This would only apply to foreign research reactor spent nuclear fuel if the Resource Conservation and Recovery Act would apply to storage and treatment of foreign research reactor spent nuclear fuel.

National Historic Preservation Act, as amended (16 USC §470 et seq.)

The National Historic Preservation Act, as amended, provides that sites with significant national historic value be placed on the *National Register of Historic Places*. There are no permits or certifications required under the Act. However, if a particular Federal activity may impact a historic property resource, consultation with the Advisory Council on Historic Preservation will usually generate a Memorandum of Agreement, including stipulations that must be followed to minimize adverse impacts. Coordination with the State Historic Preservation officer is also undertaken to ensure that potentially significant sites are properly identified and appropriate mitigative actions are implemented.

Archaeological Resource Protection Act, as amended (16 USC §470aa et seq.)

This Act requires a permit for any excavation or removal of archaeological resources from public or Native American lands. Excavations must be undertaken for the purpose of furthering archaeological knowledge in the public interest, and resources removed are to remain the property of the United States. Consent must be obtained from the Indian Tribe owning lands on which a resource is located before a permit is issued, and the permit must contain terms or conditions requested by the Tribe.

Native American Grave Protection and Repatriation Act of 1990 (25 USC §3001)

This law directs the Secretary of Interior to assume responsibilities for repatriation of Federal archaeological collections and collections held by museums receiving Federal funding that are culturally affiliated with Native American Tribes. Major actions to be taken under this law include (a) establishing a review committee with monitoring and policy-making responsibilities, (b) developing regulations for repatriation, including procedures for identifying lineal descent or cultural affiliation needed for claims, (c) oversight of museum programs designed to meet the inventory requirements and deadlines of this law, and (d) developing procedures to handle unexpected discoveries of graves or grave goods during activities on Federal or tribal land.

American Indian Religious Freedom Act of 1978 (42 USC §1996)

This Act reaffirms Native American religious freedom under the First Amendment, and sets U.S. policy to protect and preserve the inherent and constitutional right of Native Americans to believe, express, and exercise their traditional religions. The Act requires that Federal actions avoid interfering with access to sacred locations and traditional resources that are integral to the practice of religions.

Religious Freedom Restoration Act of 1993 (42 USC §2000bb et seq.)

This Act prohibits the Government, including Federal Departments, from substantially burdening the exercise of religion unless the Government demonstrates a compelling governmental interest, and the action furthers a compelling Government interest and is the least restrictive means of furthering that interest.

Endangered Species Act, as amended (16 USC §1531 et seq.)

The Endangered Species Act, as amended, is intended to prevent the further decline of endangered and threatened species and to restore these species and their habitats. The Act is jointly administered by the United States Departments of Commerce and the Interior. Section 7 of the Act requires consultation with the U.S. Fish and Wildlife Service to determine whether endangered and threatened species or their critical habitats are known to be in the vicinity of the proposed action. The Idaho National Engineering Laboratory has commenced the consultation process with the U.S. Fish and Wildlife Service (DOE, 1995c). The Savannah River Site, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site have also commenced consultations with the U.S. Fish and Wildlife Service.

Migratory Bird Treaty Act, as amended (16 USC §703 et seq.)

The Migratory Bird Treaty Act, as amended, is intended to protect birds that have common migration patterns between the United States and Canada, Mexico, Japan, and Russia. It regulates the harvest of migratory birds by specifying things such as the mode of harvest, hunting seasons, and bag limits. The Act stipulates that it is unlawful at any time, by any means, or in any manner to "kill . . . any migratory bird." Although no permit for this project is required under the Act, DOE is required to consult with the U.S. Fish and Wildlife Service regarding impacts to migratory birds and to evaluate ways to avoid or minimize these effects in accordance with the U.S. Fish and Wildlife Service Mitigation Policy.

Bald and Golden Eagle Protection Act, as amended (16 USC §668-668d)

The Bald and Golden Eagle Protection Act makes it unlawful to take, pursue, molest, or disturb bald (American) and golden eagles, their nests, or their eggs anywhere in the United States (Sections 668, 668c). A permit must be obtained from the U.S. Department of the Interior to relocate a nest that interferes with resource development or recovery operations.

Wild and Scenic Rivers Act, as amended (16 USC 1271 et seq. 71:8301 et seq.)

The Wild and Scenic Rivers Act, as amended, protects certain selected rivers of the Nation that possess outstanding scenic, recreational, geological, fish and wildlife, historical, cultural, or other similar values. These rivers are to be preserved in a free-flowing condition to protect water quality and other vital national conservation purposes. The purpose of the Act is to institute a national wild and scenic rivers system, to designate the initial rivers that are a part of that system, and to develop standards for the addition of new rivers in the future.

Occupational Safety and Health Act of 1970, as amended (29 USC §651 et seq.)

The Occupational Safety and Health Act establishes standards to enhance safe and healthful working conditions in places of employment throughout the United States. The Act is administered and enforced by the Occupational Safety and Health Administration, a U.S. Department of Labor agency. While the Occupational Safety and Health Administration and Environmental Protection Agency both have a mandate to reduce exposures to toxic substances, the Occupational Safety and Health Administration's jurisdiction is limited to safety and health conditions that exist in the workplace environment. In general, under the Act, it is the duty of each employer to furnish all employees a place of employment free of recognized hazards likely to cause death or serious physical harm. Employees have a duty to comply with the occupational safety and health standards and all rules, regulations, and orders issued under the Act. the Occupational Safety and Health Administration regulations (29 CFR) establish specific standards telling employers what must be done to achieve a safe and healthful working environment. DOE places emphasis on compliance with these regulations at its facilities and prescribes through DOE Orders the Occupational Safety and Health Act standards that contractors shall meet, as applicable to their work at Government-owned, contractor-operated facilities (DOE Order 5480.1B, 5483.1A). DOE keeps and makes available the various records of minor illnesses, injuries, and work-related deaths as required by the Occupational Safety and Health Administration regulations.

Noise Control Act of 1972, as amended (42 USC §4901 et seq.)

Section 4 of the Noise Control Act of 1972, as amended, directs all Federal agencies to carry out "to the fullest extent within their authority" programs within their jurisdictions in a manner that furthers a national policy of promoting an environment free from noise that jeopardizes health and welfare.

5.2.2 Executive Orders

Executive Order 11514 (Protection and Enhancement of Environmental Quality)

Executive Order 11514 requires Federal agencies to continually monitor and control their activities to protect and enhance the quality of the environment and to develop procedures to ensure the fullest practicable provision of timely public information and understanding of the Federal plans and programs with environmental impact to obtain the views of interested parties. The DOE has issued regulations (10 CFR 1021) and DOE Order 5440.1E for compliance with this Executive Order.

Executive Order 11988 (Floodplain Management)

Executive Order 11988 requires Federal agencies to establish procedures to ensure that the potential effects of flood hazards and floodplain management are considered for any action undertaken in a floodplain and that floodplain impacts be avoided to the extent practicable.

Executive Order 11990 (Protection of Wetlands)

Executive Order 11990 requires Governmental agencies to avoid any short- and long-term adverse impacts on wetlands wherever there is a practicable alternative.

Executive Order 12856 (Right-to-Know Laws and Pollution Prevention Requirements)

Executive Order 12856 requires all Federal agencies to reduce the toxic chemicals entering any waste stream. This order also requires Federal agencies to report toxic chemicals entering waste streams; improve emergency planning, response, and accident notification; and encourage clean technologies and testing of innovative prevention technologies.

Executive Order 12898 (Environmental Justice)

Executive Order 12898 requires Federal agencies to identify and address disproportionately high and adverse human health or environmental effects of its programs, policies, and activities on minority and low-income populations.

Table 5-2 DOE Orders Relevant to the DOE Spent Nuclear Fuel Management Program

DOE Order	Subject
1300.2A	Department of Energy Technical Standards Program (5-19-92)
1360.2B	Unclassified Computer Security Program (5-18-92)
1540.2	Hazardous Material Packaging for Transport-Administrative Procedures (9-30-86; Chg. 1. 12-19-88)
3790.1B	Federal Employee Occupational Safety and Health Program (1-7-93)
4330.4A	Maintenance Management Program (10-17-90)
4700.1	Project Management System (3-6-87)
5000.3B	Occurrence Reporting and Utilization of Operations Information (4-9-92)
5400.1	General Environmental Protection Program (11-9-88; Chg. 1, 6-29-90)
5400.2A	Environmental Compliance Issue Coordination (Errata 1-31-89)
5400.4	Comprehensive Environmental Response, Compensation, and Liability Act Requirements (10-6-89)
5400.5	Radiation Protection of the Public and the Environment (2-8-90; Chg. 2, 1-7-93)
5440.1E	National Environmental Policy Act Compliance Program (11-10-92)
5480.1B	Environmental, Safety and Health Program for DOE Operations (9-23-86; Chg. 4, 3-27-90)
5480.3	Environmental Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes (7-9-85)
5480.4	Environmental Protection, Safety, and Health Protection Standards (5-15-84; Chg. 4, 1-7-93)
5480.6	Safety of Department of Energy-Owned Nuclear Reactors (9-23-86)
5480.7A	Fire Protection (2-17-93)
5480.8A	Contractor Occupational Medical Program (6-26-92)
5480.9	Construction Safety and Health Program (11-18-87)
5480.10	Contractor Industrial Hygiene Program (6-26-85)
5480.11	Radiation Protection for Occupational Workers (12-21-88; Chg. 2, 6-29-90)
5480.15	Department of Energy Laboratory Accreditation Program for Personnel Dosimetry (12-14-87)
5480.17	Site Safety Representatives (10-05-88))
5480.18A	Accreditation of Performance-Based Training for Category A Reactors and Nuclear Facilities (07-19-91)
5480.19	Conduct of Operations Requirements for DOE Facilities (7-9-90; Chg. 1, 5-18-92)
5480.20	Personnel Selection, Qualification, Training, and Staffing Requirements at DOE Reactor and Nonreactor Nuclear Facilities (2-20-91)
5480.21	Unreviewed Safety Questions (12-24-91)
5480.22	Technical Safety Requirements (2-25-92; Chg. 1, 9-15-92)
5480.23	Nuclear Safety Analysis Reports (4-10-92)
5480.24	Nuclear Criticality Safety (8-12-92)
5480.27	Equipment Qualification for Reactor and Nonreactor Nuclear Facilities (1-15-93)
5480.28	Natural Phenomena Hazards Mitigation (1-15-93)
5480.31	Startup and Restart of Nuclear Facilities (9-15-93)
5481.1B	Safety Analysis and Review System (9-23-86; Chg. 1, 5-19-87)
5482.1B	Environment, Safety, and Health Appraisal Program (9-23-86; Chg. 1, 11-18-91)
5483.1A	Occupational Safety and Health Program for DOE Contractor Employees at Government-Owned, Contractor-Operated Facilities (6-22-83)
5484.1	Environmental Protection, Safety, and Health Protection Information Reporting Requirements (2-21-81; Chg. 7, 10-17-90)
5500.1B	Emergency Management System (4-30-91; Chg. 1, 4-30-91)
5500.2B	Emergency Categories, Classes, and Notification and Reporting Requirements (4-30-91; Chg. 1, 2-27-92)
5500.3A	Planning and Preparedness for Operational Emergencies (4-30-91; Chg. 1, 2-27-92)
5500.4A	Public Affairs Policy and Planning Requirements for Emergencies (6-8-92)
5500.7B	Emergency Operating Records Protection Program (10-23-91)
5500.10	Emergency Readiness Assurance Program (4-30-91; Chg. 1, 2-27-92)
5530.3	Radiological Assistance Program (01-14-92; Change 1, 4-10-92)
5530.5	Federal Radiological Monitoring and Assessment Center (7-10-92)
5630.11A	Safeguards and Security Program (12-7-92)
5630.12A	Safeguards and Security Inspection and Evaluation Program (6-23-92)
5700.6C	Quality Assurance (8-21-91)
5820.2A	Radioactive Waste Management (9-26-88)
6430.1A	General Design Criteria (4-6-89)

5.2.3 DOE Regulations and Orders

Through the authority of the Atomic Energy Act, DOE is responsible for establishing a comprehensive health, safety, and environmental program for its facilities. The regulatory mechanisms through which DOE manages its facilities are the promulgation of regulations and the issuance of DOE Orders.

The DOE regulations are generally found in 10 CFR. These regulations address such areas as energy conservation, administrative requirements and procedures, nuclear safety, and classified information. For the purposes of this EIS, relevant regulations include 10 CFR Part 834, Radiation Protection of the Public and the Environment; 10 CFR Part 835, Occupational Radiation Protection; 10 CFR Part 1021, Compliance with NEPA; and 10 CFR Part 1022, Compliance with Floodplains/Wetlands Environmental Review Requirements. DOE has enacted occupational radiation protection standards to protect DOE and its contractor employees. These standards are set forth in 10 CFR Part 83b, Occupational Radiation Protection. The rules in this part establish radiation protection standards, limits, and program requirements for protecting individuals from ionizing radiation resulting from the conduct of DOE activities, including those conducted by DOE contractors. The activity may be, but is not limited to, design, construction, or operation of DOE facilities. These regulations would be in effect for the construction and operation of any facilities associated with the management of foreign research reactor spent nuclear fuel.

DOE Orders generally set forth policy and the programs and internal procedures for implementing those policies. The major DOE Orders pertaining to the eventual construction and operation of spent nuclear fuel facilities within the DOE Complex are listed in Table 5-2.

5.2.4 Nuclear Regulatory Commission (NRC) Licensing Standards

DOE is proceeding with actions to implement safe, efficient, and cost-effective interim storage of its spent nuclear fuel before final disposition. The need for interim storage has led DOE to evaluate storage technologies and alternative management strategies to provide an optimum solution to storage challenges. Several commercial storage technologies under evaluation for DOE-owned spent nuclear fuel have been licensed and regulated by NRC. In addition, DOE-owned spent nuclear fuel could eventually come under the jurisdiction of NRC if it is to be disposed of in a geological repository. Therefore, DOE is considering having any new interim storage facilities reviewed to determine whether they could meet NRC licensing standards. This approach, if implemented, would provide a testing ground for the development of the technical and administrative protocols between NRC and DOE in the event that some type of NRC regulatory oversight occurs in the future.

5.3 International Regulations

Regulations of the International Atomic Energy Agency

The International Atomic Energy Agency is an agency of the United Nations headquartered in Vienna, Austria. The International Atomic Energy Agency establishes standards for radioactive materials transportation. These are published as model regulations (Safety Series No. 6) that may be adopted by individual nations. These model regulations are regularly revised and updated. Safety Series 6 was revised in 1990 (IAEA, 1990). To the extent considered feasible, the U.S. Nuclear Regulatory Commission (NRC) and the Department of Transportation both periodically review and revise their regulations to bring them into general accord with the International Atomic Energy Agency regulations.

The emphasis of the International Atomic Energy Agency model regulations is on package integrity. To that end, packagings must be shown to survive a hypothetical accident sequence that includes impact, crush, puncture, fire, and immersion. The level of protection is defined by the nature of the contents. The intent of the regulations is to maximize the shipper's contribution to safety, and the shipper (consignor) must certify "that the contents of this consignment are properly described by name; are properly packaged, marked and labeled; and are in proper condition for transport ... " (IAEA, 1990a). The carrier is responsible for following rules for stowage and for segregation from persons.

International Maritime Organization Regulations

The International Maritime Organization publishes the International Maritime Dangerous Goods Code (IMO, 1994), which was developed to supplement the provisions of the 1960 International Convention on the Safety of Life at Sea, as amended, (IMO, 1992) to which the United States is a signatory. Included are regulations that deal with carriage of radioactive material (Class 7 materials). They are based on the International Atomic Energy Agency regulations and deal with segregation of radioactive materials packages from other dangerous goods and other aspects of stowage.

5.4 Domestic Regulations for Radioactive Material Packaging and Transportation

Hazardous and Radioactive Materials Transportation Regulations

Transportation of hazardous and radioactive materials, substances, and wastes are governed by the Department of Transportation, NRC, and the Environmental Protection Agency regulations. These regulations may be found in 49 CFR Parts 171 through 178, 49 CFR Parts 383 through 397, 10 CFR Part 71, and 40 CFR Parts 262 and 265, respectively.

Department of Transportation regulations contain requirements for identifying a material as hazardous or radioactive. These regulations interface with NRC or the Environmental Protection Agency regulations for identifying material, but the Department of Transportation hazardous material regulations govern the hazard communication (such as marking, hazard labeling, vehicle placarding, and emergency response telephone number) and shipping requirements (such as required entries on shipping papers or the Environmental Protection Agency waste manifests).

NRC regulations applicable to radioactive materials transportation are found in 10 CFR Part 71, which includes detailed packaging design requirements and package certification testing requirements. Complete documentation of design and safety analysis and results of the required testing are submitted to the NRC to certify the package for use. This certification testing involves the following components: heat, physical drop onto an unyielding surface, water submersion, puncture by dropping package onto a steel bar, and gas tightness. The recent revision of 10 CFR Part 71, issued on September 28, 1995 (60 CFR 50248), is intended primarily to bring this regulation into conformance with current International Atomic Energy Agency regulations. Revised regulations applicable to the transportation of spent nuclear fuel from foreign research reactors are essentially unchanged.

The Environmental Protection Agency regulations pertaining to hazardous waste transportation are found in 40 CFR Parts 262 and 265. These regulations address labeling and record keeping requirements, including the use of the Environmental Protection Agency waste manifest, which is the required shipping paper for transporting the Resource Conservation and Recovery Act hazardous waste.

5.4.1 NRC Packaging Certification

An NRC certificate is issued as evidence that a packaging and its contents meet applicable Federal regulations. The certificate is issued on the basis of a Safety Analysis Report on the packaging design. Type B packaging must survive certain severe hypothetical accident conditions of impact, puncture, fire, and immersion. The tests are not intended to duplicate accident environments, but rather to produce damage equivalent to extreme accidents. The complete accident sequence is described in 10 CFR, Part 71.73.

Test Sequence for Type B Packagings

The effects of the tests on a package may be evaluated either by subjecting a scale model sample package to the test or by other methods acceptable to the NRC. NRC Regulatory Guide 7.9 allows assessment of package performance by analysis, prototype testing, model testing, or comparison to a similar package. To be judged as surviving, the packaging must not exceed allowable releases defined in 10 CFR 71.51. The dose rate outside the packaging must not exceed 1 rem per hour at a distance of 1 m (3.3 ft) from the packaging surface. The first three tests must be performed on the same package in this order: drop test, puncture test, and thermal test (with an immersion test following for fissile material packagings only).

The drop test consists of a 9-m (30-ft) drop onto a flat, essentially unyielding, horizontal surface, striking the package surface in the position for which maximum damage is expected. An essentially unyielding surface is one that absorbs very little of the energy of impact, which means that the energy of impact is absorbed almost entirely by the package. Unyielding surfaces are constructed of a monolithic concrete base, reinforced by Rebar and covered with a plate of battleship armor. The puncture test consists of a 1-m (40-in) drop onto the upper end of a 15-cm (6-in) solid, vertical, cylindrical bar of mild steel mounted on an essentially unyielding surface. The top of the bar must be horizontal and its edge rounded to a radius of not more than 6 mm (0.25 in).

In the thermal test, the packaging must be exposed for not less than 30 minutes to a heat flux not less than that of a radiation environment of 800°C (1,475°F) with an emissivity coefficient of a least 0.9. The surface absorptivity must be either the value that the package may be expected to possess if exposed to a fire or 0.8, whichever is greater. When it might be significant, convective heat input must be included on the basis of still, ambient air. The packaging may not be artificially cooled after external heat input ceases, and any combustion of materials of construction must be allowed to proceed until it terminates naturally.

Fissile materials packagings for which water in leakage has not been assumed for criticality analysis must be subjected to submersion under a head of water of at least 0.9 m (3 ft) for not less than 8 hours and in the attitude for which the maximum leakage is expected. All packages must be subjected to a separate test in which an undamaged cask is submerged under a head of water of at least 15 m (50 ft) for not less than 8 hours.

Although spent fuel casks have been involved in several accidents, their integrity has never been compromised. The regulatory tests are structured to place an upper bound on the kinds of damage seen in actual severe transportation accidents. Furthermore, after completion of this series of performance qualification tests, Type B packagings are further subjected to a post-accident leak-rate performance test (10 CFR 71.51). In this test, no escape of radioactive material is allowed that exceeds an A2 amount in a week. The A2 amount of an isotope is the maximum activity of that isotope in a potentially dispersible form that is allowed to be shipped in a Type A packaging, which is nonaccident resistant. Safety Series No. 6 lists A2 values for all commonly transported isotopes.

The NRC revised 10 CFR Part 7 regulations governing the transportation of radioactive materials on September 28, 1995 (60 FR 50248). These regulations become effective on April 1, 1996 (NRC, 1995). The revised regulations conform with those of the International Atomic Energy Agency and current legislative requirements. The revised regulations affecting "Type B" casks require that a spent nuclear fuel transportation cask with activity greater than 106 curies be designed and constructed so that its undamaged containment system would withstand an external water pressure of 290 psi, or immersion in 200 meters (656 ft) of water, for a period of not less than one hour without collapse, buckling, or allowing water to leak into the cask.

The use of an essentially unyielding target makes the regulatory certification tests extremely demanding. Real targets are much more yielding. For example, a lead-shield steel cask was dropped 610 m (2,000 ft) from a helicopter onto undisturbed soil (NRC, 1977). Impact velocity was 396 km per hour (235 mph). The cask penetrated 2.4 m (8 ft) into the hard soil but suffered no measurable deformation. An identical cask dropped 9 m (30 ft) onto an essentially unyielding surface during regulatory testing suffered considerably more deformation (Jefferson and Yoshimura, 1978). More recent research has expanded the study of yielding targets (e.g., concrete surfaces) and their comparison with the regulatory surface (Gonzalez et al., 1986).

5.4.2 Transportation Regulations

To assure that the transportation cask is properly prepared for transportation, trained technicians perform numerous inspections and tests (10 CFR §71.87). These tests are designed to ensure that the cask components are properly assembled and meet leak-tightness, thermal, radiation, and contamination limits. The tests and inspections are clearly identified in the Safety Analysis Report for Packaging and/or the Certificate of Compliance for each cask. Casks can only be operated by registered users who conduct operations in accordance with documented and approved quality assurance programs meeting the requirements of the regulatory authorities. Records must be maintained that document proper cask operations in accordance with the quality requirements of 10 CFR §71.91. Reports of defects or accidental mishandling must be submitted to NRC.

Communications

Proper communication assists in assuring safe preparation and handling of transportation casks. Communication is provided by labels, markings, placarding, and shipping papers or other documents. Labels (49 CFR §172.403) applied to the cask document the contents and the amount of radiation emanating from the cask exterior (transport index). The transport index lists the ionizing radiation level (in mrem/hr) at a distance of 1 m (3.3 ft) from the cask surface.

In addition to the label requirements, markings (49 CFR Subpart D and §173.471) should be placed on the exterior of the cask to show the proper shipping name and the consignor and consignee in case the cask is separated from its original shipping documents (40 CFR §172.203). Transportation casks are required to be permanently marked with the designation "Type B," the owner's (or fabricator's) name and address, the Certificate of Compliance number, and the gross weight (10 CFR §71.83).

Placards (49 CFR §172.500) are applied to the transport vehicle or freight container holding the transportation cask. The placards indicate the radioactive nature of the contents. In the United States, spent nuclear fuel is a Highway Route Controlled Quantity which must be placarded according to 49 CFR §172.507. Placards provide the first responders to a traffic or transportation accident with initial information about the nature of the contents.

Shipping papers should have entries identifying the following: the name of the shipper, emergency response telephone number, description of spent nuclear fuel, and the shipper's certificate as described in 49 CFR §172 Subpart C.

In addition, drivers of motor vehicles transporting spent nuclear fuel must have training in accordance with the requirements of 49 CFR §172.700. The training requirements include: familiarization with the regulations, emergency response information, and the spent nuclear fuel communication programs required by the Occupational Safety and Health Administration. Drivers are also required to have training on the procedures necessary for safe operation of the vehicle used to transport the spent nuclear fuel.

Except for exclusive-use shipments, requirements relating to transport indexes state that:

> ". . . the number of freight containers with packages of radioactive materials contained therein must be limited so that the total sum of the transport indexes in the containers in any hold or defined deck area does not exceed 200, and:
>
> (I) The sum of transport indexes for any individual freight container, or group of freight containers, does not exceed 50; and,
>
> (2) Each freight container or group of freight containers is (are) handled and stowed in such a manner that groups are separated from each other by a distance of at least six m (20 ft)," [49 CFR §176.704(c)].

Section 176.76(a) includes provision for freight containers with hazardous materials to be carried onboard a vessel in accordance with the following:

> " (1) The material must be in proper condition for transportation according to the requirements of this subchapter;
>
> (2) All packages in the transport vehicle or container must be secured to prevent movement in any direction. However, vertical restraint is not required if the shape of the packages and the stuffing pattern precludes shifting of the load;
>
> (3) Bulkheads made of dunnage which extend to the level of the cargo must be provided unless the packages are stowed flush with the sides or ends;
>
> (4) Dunnage must be secured to the floor when the cargo consists of dense materials or heavy packages."

Each freight container must be placarded as required by 49 CFR §172 Subpart F of the Hazardous Materials Regulations [49 CFR 176.76(f)].

Section 49 CFR 176.80 requires that radioactive materials be segregated from other hazardous materials so that they do not interact dangerously in an accident, or alternatively, requires that the radioactive material be in separate holds when stored under deck. In 49 CFR 176.83(b), a table is provided (Table II) that specifies the minimum separation distances for different classes of hazardous materials onboard a vessel. A minimum horizontal separation distance of 3 m (10 ft) projected vertically from the reference package is required. For specified hazardous materials, the "separate from" requirement means that the materials must be placed in separate holds when stowed under deck.

Marine Transport

Relevant regulations applying to transport of spent nuclear fuel by vessel are found in 10 CFR Parts 71 and 73, and 49 CFR Part 176. The USCG, part of the Department of Transportation, inspects vessels for compliance with applicable regulations and requires 24-hour prenotification (33 CFR 160.207, 211, and 213).

Section 49 CFR 171.12 (d) states that: "Radioactive materials being imported into or exported from the United States, or passing through the United States in the course of being shipped between places outside the U.S., may be offered and accepted for transportation when packaged, marked, labeled, and otherwise prepared for shipment in accordance with the IAEA 'Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6, 1985 Edition' including 'Supplement 1988.'" Certain specified conditions of this section must be complied with. For example, highway-route-controlled quantities of radioactive material must be shipped in accordance with appropriate provisions of the hazardous materials regulations and a Certificate of Competent Authority must be obtained, with any necessary revalidations. A Certificate of Competent Authority fulfills the International Atomic Energy Agency requirement for multilateral approval for a shipment of Type B packages in international commerce (IAEA, 1990a).

Section 49 CFR 176.5 details the application of the regulations to vessels: "...this subchapter applies to each domestic or foreign vessel when in the navigable waters of the United States, regardless of its character, tonnage, size or service, and whether self-propelled or not, whether arriving or departing, underway, moored, anchored, aground, or while in drydock." Exempted from the regulations are vessels not engaged in commercial service, a vessel used exclusively for pleasure, a vessel of 500 gross tons or smaller, engaged in fisheries, etc. Section 49 CFR 176.15 provides for enforcement of 40 CFR Subchapter C:

> "(a) An enforcement officer of the U.S. Coast Guard may at any time and at any place, within the jurisdiction of the United States, board any vessel for the purpose of enforcement of this subchapter and inspect any shipment of hazardous materials as defined in this subchapter."

Provision is also made in this section to detain a vessel that is in violation of the hazardous materials regulations.

The USCG may accept a certificate of loading issued by the National Cargo Bureau, Inc., as evidence that the cargo is stowed in conformity with law and regulatory requirements. The National Cargo Bureau, Inc., is a non-profit organization directed by government and industry representatives (49 CFR 176.18 authorizes inspectors of the National Cargo Bureau, Inc., to assist the USCG in administering the hazardous materials regulations). Their functions are as follows:

> "(1) Inspection of vessels for suitability for loading hazardous materials; (2) Examination of stowage of hazardous materials; (3) Making recommendations for stowage requirements of hazardous materials cargo; and, (4) Issuance of certificates of loading setting forth that the stowage of hazardous materials is in accordance with the requirements of 46 U.S.C. 170 and its subchapter."

Detailed requirements for shipping radioactive material are located in Part 176 Subpart M of the hazardous materials regulations. General radioactive materials stowage requirements of 49 CFR 176.700 state that: "(b) A package of radioactive materials which in still air has a surface temperature more than 5°C (9°F) above the ambient air may not be overstowed with any other cargo. If the package is stowed under the deck, the hold or compartment in which it is stowed must be ventilated."

Except for exclusive-use shipments, requirements of 176.704 (c) relating to transport indexes state that:

> "the number of freight containers with packages of radioactive materials contained therein must be limited so that the total sum of the transport indexes in the containers in any hold or defined deck area does not exceed 200, and: (1) The sum of transport indexes for any individual freight container, or group of freight containers, does not exceed 50; and, (2) Each freight container or group of freight containers is handled and stowed in such a manner that groups are separated from each other by a distance of at least six meters (20 feet)."

Section 176.76(a) includes provision for freight containers with hazardous materials to be carried on board a vessel in accordance with the following:

> "(1) The material must be in proper condition for transportation according to the requirements of this subchapter; (2) All packages in the transport vehicle or container must be secured to prevent movement in any direction. Vertical restraint is not required if the shape of the packages, loading pattern, and horizontal restraint preclude vertical movement of the load within the freight container or transport vehicle; (3) Bulkheads made of dunnage which extend to the level of the cargo must be provided unless the packages are stowed flush with the sides or ends; (4) Dunnage must be secured to the floor when the cargo consists of dense materials or heavy packages."

Each freight container must be placarded as required by Subpart F of Part 172 of the hazardous materials transportation regulations [49 CFR 176.76(f)].

Section 49 CFR 176.80 requires that radioactive materials be segregated from other hazardous materials so that they do not interact dangerously in an accident or, alternatively, requires that the radioactive material be in separate holds when stored under deck. In 49 CFR 176.83(b), a table is provided (Table II) that specifies the minimum separation distances for different classes of hazardous materials on board a vessel. A minimum horizontal separation distance of 3 m (10 ft) projected vertically from the reference package is required. For specified hazardous materials, the "separate from" requirement means that the materials must be placed in separate holds when stowed under deck.

Ground Transport

Overland shipments (by rail car or by truck) are regulated by a variety of the Department of Transportation and NRC regulations dealing with packaging, notification, escorts and communication. In addition, there are specific regulations for carriage by truck and carriage by rail.

When provisions are made to secure a package so that its position within the transport vehicle remains fixed during transport, with no loading or unloading between the beginning and end of transport, a package shipped overland in exclusive-use closed transport vehicles may not exceed the following radiation levels as provided in 49 CFR 173.441(b):

1. 200 millirem per hour on the external surface of the package unless the following conditions are met, in which case the limit is 1,000 millirem per hour;

    i. The shipment is made in a closed transport vehicle;

    ii. The package is secured within the vehicle so that its position remains fixed during transportation; and

    iii. There are no loading or unloading operations between the beginning and end of the transportation;

2. 200 millirem per hour at any point on the outer surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load (or enclosure is used), and on the lower external surface of the vehicle;

3. 10 millirem per hour at any point 2 m (6.6 ft) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 m (6.6 ft) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle); and

4. 2 millirem per hour in any normally occupied space.

The shipper of record must comply with the requirements of 10 CFR 71.5 and 73.37. Section 71.5 provides that all overland shipments must be in compliance with Department of Transportation and NRC regulations, which provide for security of irradiated reactor fuel. General requirements include:

- Provide notification to NRC in advance of each shipment,
- Develop a shipping plan,
- Provide escort instructions
- Establish a communication center to be staffed 24 hours a day,
- Make arrangements with local law enforcement agencies along the route for their response, if not using law enforcement personnel as escort, ensure that the escorts are trained in accordance with 10 CFR 73.37 Appendix D, and
- Ensure that escorts make notification calls every 2 hours to the communications center.

Additional requirements include having two armed escorts within heavily populated areas (when not in heavily populated areas, only one escort is needed) and the capability of communicating with the communications center and local law enforcement agencies through a radiotelephone or other NRC-approved means of two-way voice communications.

The shipper of record, as required by 49 CFR 173.22, provides physical security measures for spent fuel shipments equivalent to those of the NRC. The shipper and his agent will provide notification for unclassified spent fuel shipments to State officials.

For carriage by truck, the carrier will use interstate highways or State-designated preferred routes for movement of radioactive materials in conformity with the Department of Transportation rule known as Docket HM-164. These 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 fuel shipments constitute such quantities. Department of Transportation rules make those routes designated by appropriate State agencies enforceable by the Federal Government according to the Department of Transportation's own determination that such route designations, when accompanied by an adequate safety analysis, are likely to result in further reduction of radiological risk.

For carriage by rail car, each shipment by the railroad must comply with 49 CFR 174, in particular, 174 Subpart K, Detailed Requirements for Radioactive Materials.

5.5 Emergency Management and Response

5.5.1 Authorities and Directives

Emergency Planning and Community Right-to-Know Act of 1986 (42 USC §11001 et seq.) (also known as "SARA Title III")

Under Subtitle A of this Act, Federal facilities, including those owned by DOE, provide various information (such as inventories of specific chemicals used or stored and releases that occur from these sites) to the State Emergency Response Commission and to the Local Emergency Planning Committee to ensure that emergency plans are sufficient to respond to unplanned releases of hazardous substances. Implementation of the provisions of this Act began voluntarily in 1987, and inventory and annual emissions reporting began in 1988 based on 1987 activities and information. DOE also requires compliance with Title III as a matter of Agency policy. The requirements for this Act were promulgated by the Environmental Protection Agency in 40 CFR Parts 350 through 372.

The Toxic Substances Control Act also regulates the treatment, storage, and disposal of certain toxic substances not regulated by the Resource Conservation and Recovery Act or other statutes, particularly polychlorinated biphenyls, chlorofluorocarbons, and asbestos.

Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release (10 CFR Part 30.72 Schedule C)

This list is the basis for both the public and private sector to determine if the radiological materials they deal with must have an emergency response plan for unscheduled releases. It is one of the threshold criteria documents for DOE Hazards Assessments required by DOE Order 5500.3A, "Planning and Preparedness for Operational Emergencies" (DOE, 1991c).

Occupational Safety and Health Administration Emergency Response, Hazardous Waste Operations and Worker Right to Know (29 CFR)

This regulation sets down the Occupational Safety and Health Administration requirements for employee safety in a variety of working environments. It addresses employee emergency and fire prevention plans (Section 1910.38), hazardous waste operations and emergency response (Section 1910.120), and hazards communication (Section 1910.1200) that enables employees to be aware of the dangers they face from hazardous materials at their workplace.

Emergency Management and Assistance (44 CFR 1.1)

This regulation contains the policies and procedures for the Federal Emergency Management Act, National Flood Insurance Program, Federal Crime Insurance Program, Fire Prevention and Control Program, Disaster Assistance Program, and Preparedness Program including radiological planning and preparedness.

Hazardous Materials Tables & Communications, Emergency Response Information Requirements (49 CFR Part 172)

The regulatory requirements for marking, labeling, placarding, and documenting hazardous materials shipments are defined in this regulation. It also specifies the requirements for providing hazardous material information and training.

Public Law 93-288, as Amended by Public Law 100-707, "Robert T. Stafford Disaster Relief and Emergency Assistance Act," November 23, 1988

The Robert T. Stafford Disaster Relief and Emergency Assistance Act, P.L. 93-288, as amended, provides an orderly and continuing means of assistance by the Federal Government to State and local governments in carrying out their responsibilities to alleviate the suffering and damage resulting from disasters. The President, in response to a State Governor's request, may declare an "emergency" or "major disaster," in order to provide Federal assistance under the Act. The President, in Executive Order 12148, delegated all functions, except those in Sections 301, 401, and 409, to the Director, Federal Emergency Management Agency. The Act provides for the appointment of a Federal Coordinating Officer who will operate in the designated area with a State Coordinating Officer for the purpose of coordinating State and local disaster assistance efforts with those of the Federal Government.

Public Law 96-510, "Comprehensive Environmental Response, Compensation, and Liability Act of 1980," Section 104(i), 42 U.S.C. 9604(i)

More popularly known as "Superfund," this Act provides the needed general authority for Federal and State governments to respond directly to hazardous substances incidents. The Act requires reporting of spills, including radioactive, to the National Response Center.

Public Law 98-473, Justice Assistance Act of 1984

These Department of Justice regulations implement the Emergency Federal Law Enforcement Assistance functions vested in the Attorney General. Those functions were established to assist State and/or local units of government in responding to a law enforcement emergency. The Act defines the term "law enforcement emergency" as an uncommon situation which requires law enforcement, which is or threatens to become of serious or epidemic proportions, and with respect to which State and local resources are inadequate to protect the lives and property of citizens, or to enforce the criminal law. Emergencies that are not of an ongoing or chronic nature, such as the Mount Saint Helens volcanic eruption, are eligible for Federal law enforcement assistance. Such assistance is defined as funds, equipment, training, intelligence information, and personnel. Requests for assistance must be submitted in writing to the Attorney General by the chief executive office of a State. The Plan does not cover the provision of law enforcement assistance. Such assistance will be provided in accordance with the regulations referred to in this paragraph [28 CFR Part 65, implementing the Justice Assistance Act of 1984] or pursuant to any other applicable authority of the Department of Justice.

Communications Act of 1934, as Amended

This Act gives the Federal Communications Commission emergency authority to grant Special Temporary Authority on an expedited basis to operate radio frequency devices.

5.5.2 Executive Orders

Executive Order 10480, as Amended, "Further Providing for the Administration of the Defense Mobilization Program," August 1953

Part II of the Order delegates to the Director, Federal Emergency Management Agency, with authority to redelegate, the priorities and allocation functions conferred on the President by Title I of the Defense Production Act of 1950, as amended.

Executive Order 12148, "Federal Emergency Management," July 20, 1979

Executive Order 12148 transferred functions and responsibilities associated with Federal emergency management to the Director, Federal Emergency Management Agency. The Order assigns the Director, Federal Emergency Management Agency, the responsibility to establish Federal policies for and to coordinate all civil defense and civil emergency planning, management, mitigation, and assistance functions of Executive Agencies.

Executive Order 12472, "Assignment of National Security and Emergency Preparedness Telecommunications Functions," April 3, 1984

Executive Order 12472 establishes the National Communication System. The National Communication System consists of the telecommunications assets of the entities represented on the National Communication System Committee of Principals and an administrative structure consisting of the Executive Agent, the National Communication System Committee of Principals, and the Manager. The National Communication System Committee of Principals consists of representatives from those Federal departments, agencies, or entities, designated by the President, which lease or own telecommunications facilities or services of significance to national security or emergency preparedness.

Executive Order 12656, "Assignment of Emergency Preparedness Responsibilities," November, 1988

This order assigns emergency preparedness responsibilities to Federal departments and agencies.

5.5.3 Emergency Planning Documents

"Federal Radiological Emergency Response Plan," November 1985

This document is to be used by Federal agencies in peacetime radiological emergencies. It primarily concerns the off-site Federal response in support of State and local governments with jurisdiction for the emergency. The Federal Radiological Emergency Response Plan provides the Federal Government's concept of operations based on specific authorities for responding to radiological emergencies, outlines Federal policies and planning assumptions that underlie this concept of operations and on which Federal agency response plans were based, and specifies authorities and responsibilities of each Federal agency that may have a significant role in such emergencies.

"National Plan for Telecommunications Support [in Non-Wartime Emergencies," January 1992

This plan provides guidance in planning for and providing telecommunications support for Federal agencies involved in emergencies, major disasters, and other urgent events, excluding war.

Department of Defense Directive 3025.1, "Military Support to Civil Authorities," 1992

This directive outlines Department of Defense policy on assistance to the civilian sector during disasters and other emergencies. Use of the Department of Defense military resources in civil emergency relief operations will be limited to those resources not immediately required for the execution of the primary defense mission. Normally, the Department of Defense military resources will be committed as a supplement to non-Department of Defense resources that are required to cope with the humanitarian and property protection requirement caused by the emergency. In any emergency, commanders are authorized to employ Department of Defense resources to save lives, prevent human suffering, or mitigate great property loss. Upon declaration of a major disaster under the provisions of P.L. 93-288, as amended, the Secretary of the Army is the Department of Defense Executive Agent, and the Director of Military Support is the action agent for civil emergency relief operations. Military personnel will be under command of and directly responsible to their military superiors and will not be used to enforce or execute civil law in violation of 18 U.S.C. 1385, except as otherwise authorized by law. Military resources shall not be procured, stockpiled, or developed solely to provide assistance to civil authorities during emergencies.

Federal Preparedness Circular 8, "Public Affairs in Emergencies"

This Circular establishes the Interagency Committee on Public Affairs in Emergencies to coordinate public information planning and operations for management of emergency information. The Circular was reviewed in draft by the Interagency Committee on Public Affairs in Emergencies and will receive formal department and agency review.

American Red Cross Disaster Services Regulations and Procedures, ARC 3003, January 1984

This document details the delegation of disaster services program responsibilities to officials and units of the American Red Cross. Also defined are the American Red Cross administrative regulations and procedures for disaster planning, preparedness, and response.

Statement of Understanding between the Federal Emergency Management Agency and the American National Red Cross, January 22, 1982

The statement of understanding between the Federal Emergency Management Act and the American National Red Cross describes major responsibilities in disaster preparedness planning and operations in the event of a war-caused national emergency or a peacetime disaster, outlines areas of mutual support and cooperation, and provides a frame of reference for similar cooperative agreements between State and local governments and the operations headquarters and chapters of the American Red Cross.

6. List of Preparers

Name:	Charles R. Head
Affiliation:	Department of Energy
Education:	MS, Control Theory, George Washington University BS, Electrical Engineering, Rice University
Experience/ Technical Specialty:	Twenty-seven years. Nuclear safety oversight, safeguards and security requirements, Tiger Team assessments, spent fuel storage.
EIS Responsibility:	DOE Foreign Research Reactor Spent Nuclear Fuel Project Manager
Name:	Patrick J. Wells
Affiliation:	Department of Energy
Education:	MS, Engineering Management, George Washington University BS, Civil Engineering, Marquette University
Experience/ Technical Specialty:	Eleven years. Occupational safety and health, oversight system acquisition, reliability, and maintainability.
EIS Responsibility:	Assistant to the DOE Project Manager
Name:	Darren W. Piccirillo
Affiliation:	Department of Energy
Education:	BS, Physics, Southern Connecticut State University
Experience/ Technical Specialty:	Eight years. Reactor plant construction and operation.
EIS Responsibility:	Spent fuel characterization, storage impact analysis
Name:	Eleanor R. Busick
Affiliation:	Department of State
Education:	MA, Economics, Yale University BA, Oberlin College
Experience/ Technical Specialty:	Nineteen years. Nuclear nonproliferation policy, economics (fuel markets, finance).
EIS Responsibility:	Nonproliferation and national policies

Name:	Fred McGoldrick
Affiliation:	Department of State
Education:	PhD, Political Science, American University BA, Boston College
Experience/ Technical Specialty:	Twenty-three years. Nuclear nonproliferation policy.
EIS Responsibility:	Nonproliferation and national policies

Name:	Ibrahim H. Zeitoun
Affiliation:	Science Applications International Corporation
Education:	PhD, Environmental Sciences, Michigan State University MS, Fisheries, Michigan State University BS, Chemistry & Zoology, University of Alexandria
Experience/ Technical Specialty:	Twenty-two years. Waste management, environmental assessment, and NEPA compliance.
EIS Responsibility:	Overall EIS Project Manager. NEPA compliance, spent fuel management environmental justice

Name:	Audrey J. Adamson
Affiliation:	Science Applications International Corporation
Education:	BA, University of Michigan MPA, George Washington University
Experience/ Technical Specialty:	Eleven years. International, environmental, and technology policy analysis, public outreach, and technical writing.
EIS Responsibility:	Emergency response and communication planning

Name:	Douglas H. Amick
Affiliation:	Science Applications International Corporation
Education:	AS, Law Enforcement, Walter State Community College
Experience/ Technical Specialty:	Twenty years. Marine ports, port safety inspections and security, plant operations, transportation risks.
EIS Responsibility:	Port safety, operation and characteristics

Name:	Paula W. Austin
Affiliation:	Science Applications International Corporation
Education:	BS, Management and Technology, University of Maryland
Experience/ Technical Specialty:	Nineteen years. Nuclear and environmental policy analysis, public outreach, technical writing.
EIS Responsibility:	EIS Summary, public hearings task leader

Name:	Rakesh Bahadur
Affiliation:	Science Applications International Corporation
Education:	PhD, Groundwater Hydrology, Colorado State University MS, Groundwater Hydrology, Colorado State University MSc, Geology, Punjab University
Experience/ Technical Specialty:	Fifteen years. Hydrology, site characterization, environmental assessment, and risk assessment.
EIS Responsibility:	Affected environment, Geographic Information Systems, Environmental Justice

Name:	Bruce M. Biwer
Affiliation:	Argonne National Laboratory
Education:	PhD, Chemistry, Princeton University MA, Chemistry, Princeton University BA, Chemistry, St. Anselm College
Experience/ Technical Specialty:	Fourteen years. Radiological pathway analysis, dose calculations, radiological transportation risk analysis.
EIS Responsibility:	Radiological transportation risk analysis

Name:	Keith R. Brown
Affiliation:	Science Applications International Corporation
Education:	MS, Mathematics, Idaho Sate University BS, Mathematics, Idaho State University
Experience/ Technical Speciality:	Thirty-four years. Transportation casks, research reactor operations, and spent nuclear fuel management
EIS Responsibility:	Spent fuel management, transportation assessment

Name:	Burrus M. Carnahan
Affiliation:	Science Applications International Corporation
Education:	LLM, International Law, University of Michigan JD, Northwestern University BA, Political Science, Drake University
Experience/ Technical Specialty:	Twenty-one years. Weapons proliferation, international agreements, and arms control.
EIS Responsibility:	Nonproliferation policies
Name:	Harry Chernoff
Affiliation:	Science Applications International Corporation
Education:	MBA, Marymount University BA, Economics, College of William & Mary
Experience/ Technical Specialty:	Seventeen years. Economics, socioeconomics, finance, and engineering economics.
EIS Responsibility:	Costs, and Socioeconomics
Name:	Louis Cofone, Jr.
Affiliation:	Science Application International Corporation
Education:	MS, Microbiology, University of Dayton BS, Biology, Philadelphia College of Pharmacy and Science
Experience/ Technical Specialty:	Twenty years. Computer information analysis, data base development and management, environmental and health risk assessment.
EIS Responsibility:	Development and management of the comment tracking and document control system
Name:	Cecil C. Cross, III
Affiliation:	Science Applications International Corporation
Education:	MEM, Environmental Management, Duke University BA, Biology, Gettysburg College
Experience/ Technical Specialty:	Thirteen years. Regulatory compliance, environmental assessment, and hazardous and mixed waste management.
EIS Responsibility:	Port operation, port environmental and climatic conditions

Name:	Larry Danese
Affiliation:	Science Applications International Corporation
Education:	MBA, Florida International University BS, Electrical Engineering, University of Florida
Experience/ Technical Specialty:	Twenty-three years. Cask design, transportation systems, emergency response, and regulatory compliance.
EIS Responsibility:	Transportation casks and regulations, port operation regulations and activities

Name:	Gary M. DeMoss
Affiliation:	Science Applications International Corporation
Education:	MS, Engineering Administration, Virginia Polytechnic Institute BS, Mechanical Engineering, University of Virginia
Experience/ Technical Specialty:	Thirteen years. Risk analysis, reliability and safety engineering, uranium enrichment, and transportation.
EIS Responsibility:	Ground transportation safety and impact analysis, port selection and operation

Name:	Scott E. Drummond, Jr.
Affiliation:	Science Applications International Corporation
Education:	BS, Marine Transportation, SUNY Maritime College
Experience/ Technical Specialty:	Forty-two years. Strategic sealift, logistics support, ocean survey, nautical charting, SWATH ship design and operation.
EIS Responsibility:	Port information

Name:	Habib A. Durrani
Affiliation:	Science Applications International Corporation
Education:	BSc, Engineering Science, Peshawar University
Experience/ Technical Specialty:	Twenty years. Nuclear facilities operation, design maintenance regulations and safety
EIS Responsibility:	Chemical separation technologies

Name:	Barbara M. Ebert
Affiliation:	Science Applications International Corporation
Education:	MA, National Security Studies, Georgetown University BS, Foreign Service, Comparative and Regional Studies, Georgetown University
Experience/ Technical Specialty:	Twelve years. Weapons proliferation.
EIS Responsibility:	Nonproliferation policies

Name:	Martin W. Ebert
Affiliation:	Science Applications International Corporation
Education:	BSc, Nuclear Engineering, University of Arizona MSc, Applied Physics, University of Strathclyde
Experience/ Technical Specialty:	Twenty-four years. Nuclear powerplant operations, spent fuel technology, and technical safety requirements.
EIS Responsibility:	Marine and port safety and impact analysis, public hearings response coordinator

Name:	Daniel W. Gallagher
Affiliation:	Science Applications International Corporation
Education:	MS, Nuclear Engineering, Rensselaer Polytechnic Institute BS, Nuclear Engineering, Rensselaer Polytechnic Institute
Experience/ Technical Specialty:	Fifteen years. Reliability and risk engineering, probabilistic safety assessment, plant design, and regulatory analysis.
EIS Responsibility:	Marine and port safety and impact analysis

Name:	Reginald L. Gotchy
Affiliation:	Science Applications International Corporation
Education:	PhD, Radiation Biology, Colorado State University MS, Radiation Health, Colorado State University BS, Zoology, University of Washington
Experience/ Technical Specialty:	Twenty-six years. NEPA compliance, safety analysis, risk assessment, radiation biology, health physics (Certified Health Physicist), and emergency response planning.
EIS Responsibility:	Port selection and radiological consequences and health effects

Name:	Peter Grier
Affiliation:	Science Applications International Corporation
Education:	BS, Psychology, University of Maryland
Experience/ Technical Specialty:	Twenty years. Emergency management, commercial nuclear energy, quality assurance, and transportation regulatory compliance.
EIS Responsibility:	Emergency response, security, and communication planning

Name:	Timothy T. Holmes
Affiliation:	Science Applications International Corporation
Education:	JD, University of Kansas School of Law BA, Business, Washburn University
Experience/ Technical Specialty:	Four years. Legal and environmental analysis, involving NEPA, RCRA, and other environmental regulations, document review and contract compliance.
EIS Responsibility:	Comment Response Document task leader, technical editor

Name:	Joseph W. James
Affiliation:	Science Applications International Corporation
Education:	BSG, Administration of Justice, American University MA, Management, Central Michigan University PhD, Environmental Science, LaSalle University
Experience/ Technical Specialty:	Thirty years. Nuclear safeguards and security, standards development, quality assurance, regulatory analysis, and licensing support.
EIS Responsibility:	Safeguards and security planning

Name:	Roy Karimi
Affiliation:	Science Applications International Corporation
Education:	ScD, Nuclear Engineering, Massachusetts Institute of Technology NE, Nuclear Engineering, Massachusetts Institute of Technology MS, Nuclear Engineering, Massachusetts Institute of Technology BSc, Chemical Engineering, Abadan Institute of Technology
Experience/ Technical Specialty:	Fourteen years. Nuclear powerplant safety, risk and reliability analysis, design analysis, and probabilistic risk assessment.
EIS Responsibility:	Spent fuel characterization, accident and impact analysis, quality control reviews

Name:	Stephen J. Krill, Jr.
Affiliation:	Science Applications International Corporation
Education:	BS, Nuclear and Power Engineering, University of Cincinnati
Experience/ Technical Specialty:	Five years. Safety and risk analysis, reactor and fuel processing system design, operation and inspection, and emergency preparedness.
EIS Responsibility:	Transportation cask descriptions, environmental consequences

Name:	Merritt E. Langston, PE
Affiliation:	Science Applications International Corporation
Education:	BS, MS, Metallurgical Engineering, Missouri School of Mines
Experience/ Technical Specialty:	Thirty-two years. Quality management, nuclear engineering, defense programs, nuclear waste management. Six years. Reactor containment materials development.
EIS Responsibility:	Technical reviews, quality control task leader
Name:	Christi D. Leigh
Affiliation:	Sandia National Laboratories
Education:	PhD, Engineering, University of New Mexico MS, Chemical Engineering, Stanford University BS, Chemical Engineering, Arizona State University
Experience/ Technical Specialty:	Six years. Radioactive and hazardous waste management and minimization. Five years. Nuclear reactor safety.
EIS Responsibility:	At-sea submerged cask risk assessment
Name:	Charles D. Massey
Affiliation:	Sandia National Laboratories
Education:	PhD, Radiation Health, University of Pittsburgh MS, Health Physics, University of Pittsburgh BS, Marine Transportation, U.S. Merchant Marine Academy
Experience/ Technical Specialty:	Thirteen years. NEPA, risk assessment, transportation and energy technology evaluation.
EIS Responsibility:	Marine transportation risk assessment and impacts
Name:	Ronya J. McMillen
Affiliation:	Science Applications International Corporation
Education:	MA, International Science & Technology Policy, George Washington University BS, Sociology, Chatham College
Experience/ Technical Specialty:	Thirteen years. Technical analysis and report writing. Four years public outreach and policy analysis on domestic and foreign nuclear technology regulatory issues.
EIS Responsibility:	EIS Summary Coordinator, Public comment and hearing summaries

Name:	Charles D. Miller
Affiliation:	Science Applications Internal Corporation
Education:	BS, Marine Transportation, SUNY Maritime College
Experience/ Technical Specialty:	Forty years. Transportation/distribution system design, planning, implementation,.and management.
EIS Responsibilty:	Port information
Name:	Todd Miller
Affiliation:	Science Applications International Corporation
Education:	BS, Civil Engineering, Worcester Polytechnic Institute
Experience/ Technical Specialty:	Four years. Safety analysis, environmental assessment, NEPA compliance.
EIS Responsibility:	Accident analysis, radiological consequences
Name:	Steven M. Mirsky
Affiliation:	Science Applications International Corporation
Education:	MS, Nuclear Engineering, Pennsylvania State University BS, Mechanical Engineering, The Cooper Union
Experience/ Technical Specialty:	Nineteen years. Safety analysis, nuclear powerplant design, operations, and foreign nuclear powerplant system analysis.
EIS Responsibility:	Storage technology, safety and impact analysis
Name:	Frederick A. Monette
Affiliation:	Argonne National Laboratory
Education:	MS, Health Physics, Colorado State University BA, Physics, St. Johns University
Experience/ Technical Specialty:	Six years. Radiological risk assessment, radiological transportation risk analysis, dose calculations.
EIS Responsibility:	Radiological transportation risk and impacts analysis
Name:	Michael Moore
Affiliation:	Science Applications International Corporation
Education:	BA, Economics, University of Maryland
Experience/ Technical Specialty:	Twelve years. Analysis and design of environmental/waste information systems, drafting and editing of technical documents for energy, environmental, and defense initiatives.
EIS Responsibility:	Quality control reviews, technical editor

Name:	Alexander P. Murray
Affiliation:	Science Applications International Corporation
Education:	MS, Chemical Engineering, Carnegie-Mellon University BS, Chemical Engineering, Carnegie-Mellon University
Experience/ Technical Specialty:	Nineteen years. Waste management, environmental and regulatory compliance, design engineering and computer modeling, nuclear reactors and systems, dose analysis, nuclear fuel cycle, and spent fuel reprocessing.
EIS Responsibility:	Storage and chemical separation technologies

Name:	Iral C. Nelson
Affiliation:	Pacific Northwest Laboratories
Education:	MA, Physics, University of Oregon Diplomate, American Board of Health Physics BS, Mathematics, University of Oregon
Experience/ Technical Specialty:	Thirty-nine years. Health physics, radiation protection, and NEPA compliance and reviews.
EIS Responsibility:	Affected environment, environmental consequences

Name:	Aris Papadopoulos
Affiliation:	Science Applications International Corporation
Education:	MS, Nuclear Engineering, University of Utah BS, Physics, Hamline University
Experience/ Technical Specialty:	Twenty-two years. Safety analysis assessment, regulatory reviews, reactor safety, fuel cycle facility systems, radioactive waste management, and accident analysis support.
EIS Responsibility:	Transportation casks, storage alternatives, and impact assessment

Name:	Kathleen Rhoads
Affiliation:	Pacific Northwest Laboratories
Education:	MS, Radiological Sciences, University of Washington BS, Microbiology, University of Washington
Experience/ Technical Specialty:	Nineteen years. Risk assessment, radiation doses, health effects from energy production.
EIS Responsibility:	Radiological consequences analysis

Name:	Van Romero
Affiliation:	Sandia National Laboratories
Education:	PhD, Physics, State University of New York MS, Physics, New Mexico Tech BS, Physics, New Mexico Tech
Experience/ Technical Specialty:	Fifteen years. Environmental health physics and radiation protection, NEPA compliance, DOE order compliance, environmental impact testing, risk assessment, nuclear safety, health physics, radiation transport, and nuclear emergency response.
EIS Responsibility:	Radiation exposure analysis for marine transport

Name:	William B. Samuels
Affiliation:	Science Applications International Corporation
Education:	PhD, Biology, Fordham University MS, Marine Science, Long Island University BS, Biology & Geology, University of Rochester
Experience/ Technical Specialty:	Sixteen years. Geographic Information Systems, computer simulation and mathematical modeling, environmental database management systems.
EIS Responsibility:	Geographic Information Systems, environmental justice

Name:	Elizabeth C. Saris
Affiliation:	Science Applications International Corporation
Education:	BA, Political Science, George Washington University
Experience/ Technical Specialty:	Fifteen years. Energy and environmental policy analysis, public outreach, and technical writing.
EIS Responsibility:	EIS Summary, public hearings support

Name:	Patrick R. Schwab
Affiliation:	Science Applications International Corporation
Education:	PhD, Nuclear Engineering, University of Wisconsin MS, Nuclear Engineering, University of Wisconsin BS, Nuclear Engineering, Kansas State University
Experience/ Technical Specialty:	Eighteen years. Design criteria, technical safety surveys, foreign nuclear technology analysis, configuration studies, and spent fuel reprocessing.
EIS Responsibility:	Environmental and policy consequences, chemical separation technologies and impacts

Name:	Barry Smith
Affiliation:	Science Applications International Corporation
Education:	JD, George Washington University National Law Center BA, Political Science, Indiana University
Experience/ Technical Specialty:	Twenty-three years. NEPA compliance, environmental law, regulatory compliance, and waste management.
EIS Responsibility:	Environmental regulation/compliance

Name:	Jeremy L. Sprung
Affiliation:	Sandia National Laboratories
Education:	PhD, Physical-Organic Chemistry, UCLA BA, Chemistry, Yale University
Experience/ Technical Specialty:	Twenty-nine years. Photochemistry and air pollution, reactor accident consequences, reactor safety studies, and transportation risk assessment.
EIS Responsibility:	Port accident risk analysis

Name:	Donna J. Stucky
Affiliation:	Pacific Northwest Laboratories
Education:	MS, Agricultural Economics, Purdue University BA, Economics, Pacific Lutheran University
Experience/ Technical Specialty:	Two years. Economic research.
EIS Responsibility:	Environmental consequences

Name:	Robert Wayland
Affiliation:	Science Applications International Corporation
Education:	PhD, Atmospheric Science, North Carolina State University MS, Environmental Science, University of Virginia BA, Environmental Science, University of Virginia
Experience/ Technical Specialty:	Eleven years. Boundary-layer meteorology, atmospheric structure and composition, ocean-atmosphere interactions, atmospheric modeling.
EIS Responsibility:	Port meteorological data assessments, site nonradiological impact analyses

Name:	Timothy Wheeler
Affiliation:	Sandia National Laboratories
Education:	MS, Systems Engineering, University of Virginia BS, Mechanical Engineering, University of New Hampshire
Experience/ Technical Specialty:	Fourteen years. NEPA compliance, radioactive material transportation risk analysis, probabilistic risk assessment.
EIS Responsibility:	At-sea submerged cask risk assessment

Name:	John W. Williams
Affiliation:	Science Applications International Corporation
Education:	PhD, Physics, New Mexico State University MS, Physics, New Mexico State University BS, Mathematics, North Texas State University
Experience/ Technical Specialty:	Twenty years. NEPA compliance, electromagnetic models, air quality modeling, ionizing radiation impacts and safety.
EIS Responsibility:	Environmental justice, ports selection, quality control reviews

Name:	Steven E. Wujciak
Affiliation:	U.S. Department of Transportation, Research & Special Projects Administration, Volpe National Transportation System Center
Education:	MBA, Anna Maria College BS, Business Administration, Anna Maria College
Experience/ Technical Specialty:	Fifteen years. Operations research, transportation analysis, emergency preparedness.
EIS Responsibility:	Ground transportation analysis

Name:	Maron D. Wylie
Affiliation:	U.S. Department of Transportation, Research & Special Projects Administration, Volpe National Transportation System Center
Education:	MS, Math and Computer Science, Worcester State College BS, Business Administration, University of Southern Mississippi
Experience/ Technical Specialty:	Fifteen years. Operations research, transportation analysis, emergency preparedness.
EIS Responsibility:	Ground transportation analysis

Name	Michael R. Zanotti
Affiliation:	Science Applications International Corporation
Education:	MPA, Administrative Management and Organization, Golden Gate University MPA, Health Services Administration, Golden Gate University BA, Behavioral Sciences, University of Maine AA, Criminal Justice, University of Maine
Experience/ Technical Specialty:	Fifteen years. Certified Emergency Manager (CEM), Emergency management, emergency response, fire response, hazardous materials response, facilities operation.
EIS Responsibility:	Emergency management and response

7. Agencies Consulted

The following agencies were consulted in the development of this Draft Environmental Impact Statement.

Federal Agencies

Arms Control and Disarmament Agency
Military Traffic Management Command
Military Ocean Terminal, Oakland (CA)
Military Ocean Terminal, Sunny Point (NC)
Naval Weapons Station, Concord (CA)
Naval Weapons Station, Charleston (SC)
Port Hueneme (CA) Naval Construction Battalion Center
U.S. Department of Defense
U.S. Department of Army
U.S. Coast Guard
U.S. Merchant Marine Academy
U.S. Fish and Wildlife Service

State Agencies

Alabama Department of Conservation and Natural Resources
Alabama Department of Environmental Management, Water Quality Division
Alabama Natural Heritage Program
Alabama State Docks, Mobile (AL)
California Fish & Game Heritage Program
California Regional Water Quality Control Board, San Francisco Bay Region
Delaware Department of Natural Resources and Environmental Control, Division of Water Resources
Delaware Natural Heritage Inventory
Florida Department of Environmental Regulation, Bureau of Surface Water Management
Florida Natural Areas Inventory
Fort Clinch State Park, Amelia Island, FL
Georgia Department of Natural Resources, Environmental Protection Division
Georgia Department of Natural Resources, Wildlife Resources Division
Georgia Ports Authority
John U. Lloyd Beach State Recreation Area, Port Everglades, FL
Louisiana Department of Environmental Quality
Louisiana Natural Heritage Program
Maryland Natural Heritage Program
Massachusetts Port Authority
Maryland Port Administration
Mississippi Department of Environmental Quality, Water Quality Division
Mississippi Natural Heritage Program
Mississippi State Port Authority at Gulfport
New Hampshire Port Authority
New Jersey Natural Heritage Program
North Carolina Department of Environment, Health, and Natural Resources, Division of Environmental Management
North Carolina Natural Heritage Program
North Carolina State Ports Authority
Oregon Natural Heritage Program
Pennsylvania Department of Environmental Resources, Water Quality Division
Pennsylvania Natural Diversity Inventory
Ports Authority of New York & New Jersey
South Carolina Department of Health and Environmental Control, Water Quality Division
South Carolina Heritage Trust
South Carolina State Ports Authority
South Jersey Port Corporation
Virginia Department of Conservation and Recreation, Division of Natural Heritage
Virginia Department of Environmental Quality, Water Division
Virginia Water Control Board
Virginia Port Authority
Washington Department of Wildlife

Local Agencies

Bridgeport Port Authority (CT)
Commissioners of Pilotage, Port of Charleston (SC)
Jacksonville Port Authority (FL)
Manatee County Port Authority (FL)
Penn Terminals (Port of Eddystone, PA)
Port Authority of Greater New Orleans (LA)
Port Everglades Authority (FL)
Port of Albany (NY)
Port of Alexandria (VA)
Port of Baton Rouge (LA)
Port of Beaumont (TX)
Port of Corpus Christi (TX)
Port of Fall River (MA)

Local Agencies (Continued)

- Port of Fernandina (FL)
- Port of Galveston (TX)
- Port of Grays Harbor (WA)
- Port of Houston Authority (TX)
- Port of Hueneme (CA)
- Port of Long Beach (CA)
- Port of Longview (WA)
- Port of Los Angeles (CA)
- Port of Miami (FL)
- Port of New Haven (CT)
- Port of Oakland (CA)
- Port of Palm Beach (FL)
- Port of Port Arthur (TX)
- Port of Portland (ME)
- Port of Portland (OR)
- Port of Portsmouth (NH)
- Port of Richmond (CA)
- Port of Richmond Commission (VA)
- Port of San Francisco (CA)
- Port of Seattle (WA)
- Port of Tacoma (WA)
- Port of Vancouver, U.S.A. (WA)
- Port of Wilmington (DE)
- Port of Wilmington (NC)
- San Diego Unified Port District (CA)
- Tampa Port Authority (FL)

Other

- Australian Nuclear Science & Technology Organization (ANSTO)
- Austrian Research Centre, Austria
- Belgian Nuclear Research Centre
- GKSS Research Center, Germany
- Hahn-Meitner Institut Berlin, Germany
- Interfaculty Reactor Institute, Delft University of Technology, The Netherlands
- Joint Research Centre-Petten, Institute for Advanced Materials, The Netherlands
- National Center for Scientific Research, "Demokritos," Greece
- Paul Scherrer Institute, Switzerland
- RISO National Laboratory, Denmark
- Studsvik Nuclear AB, Sweden
- United Kingdom Atomic Energy Authority, Thurso, Dounreay Caithness, Scotland

8. References

AAPA (American Association of Port Authorities), 1993, *The 1993 AAPA Directory, Seaports of the Western Hemisphere*, K-111 Directory Corporation, New York, NY, January.

Abbott, M. L., J. M. Brooks and K. L. Martin, 1990, *NPR Environmental Impact at the INEL: Air Quality, Cooling Tower, and Noise*, NPRD-90-059, EG&G Idaho, Inc., Idaho Falls, ID, November.

Ackerman, D. J., 1991, *Transmissivity of the Perched Aquifers at the INEL*, WRIR 91-4114, DOE/ID-22099, U.S. Geological Survey, Idaho Falls, ID.

Aiken County, 1991, *Functional Performance Standards for Non-residual Uses, Zoning and Development Standards Ordinance*, Chapter VI, Section 24-204 - Noise, Aiken, SC.

Analysas Corporation, 1994, *TRANSCOM Shipment and Analysis of Urgent Relief Training Statistics Facsimile*, Oak Ridge, TN, September 21.

Anderson, S. R., 1991, *Stratigraphy of the Unsaturated Zone and Uppermost Part of the Snake River Plain Aquifer at the Idaho Chemical Processing Plant and Test Reactor Area*, WRIR 91-4010, DOE/ID-22095, U.S. Geological Survey, Idaho Falls, ID, January.

ANSI (American National Standards Institute), 1975, Standard N305-1975, *Design Objectives for Highly Radioactive Solid Material Handling and Storage Facilities in a Reprocessing Plant,* March.

Arnett, M. W., L. K. Karapatakis and A. R. Mamatey, 1993, *Savannah River Site Environmental Report for 1992*, WSRC-TR-93-075, Westinghouse Savannah River Company, Aiken, SC.

Arthur, W. J., J. W. Connelly, D. K. Halford and T. D. Reynolds, 1984, *Vertebrates of the Idaho National Engineering Laboratory*, DOE/ID-12099, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, ID, July.

Barraclough, J. T., B. D. Lewis and R. G. Jensen, 1981, *Hydrologic Conditions at the INEL, Idaho-Emphasis: 1974-1978*, WRI Open File Report 81-526, IDO-22060, U.S. Geological Survey, Idaho Falls, ID, April.

Beavers, J. E., W. E. Manrod and W. C. Stoddart, 1982, *Recommended Seismic Hazard Levels for the Oak Ridge, Tennessee; Paducah, Kentucky; Fernald, Ohio; and Portsmouth, Ohio, Department of Energy Reservations*, K/BD-1025/R1, Oak Ridge, TN, December.

Bendixsen, C. L., 1995, Lockheed Idaho Technologies Company, personal communication to A. Zeitoun, Science Applications International Corporation, October 18.

Bennett, C. M., 1990, *Streamflow Losses and Ground-Water Level Changes Along the Big Lost River at the INEL*, WRIR 90-4067, DOE/ID-22091, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, ID.

BNFL (British Nuclear Fuels Limited), 1994, *UK-Based Processing as a Spent Fuel Management Option for U.S. DOE*, Risley, England, May.

Bolt, B. A., 1978, *Earthquakes: A Primer*, W. H. Freeman & Co., San Francisco, CA.

Bowdle, K., 1994, U.S. Department of Energy, Environmental Restoration Division, personal communication with K. Laudkrohn, Haliburton NUS Corporation, Gaithersburg, MD, May 20.

Boyle, J. W., R. Blumbert, S. J. Cotter, G. S. Hill, C. R. Kerley, R. H. Ketelle, R. L. Kroodsma, D. W. Lee, R. C. Martin, R. D. Roop, D. N. Secora, W. P. Staub and R. E. Thoma, 1982, *Environmental Analysis of the Operation of Oak Ridge Laboratory (X-10 Site)*, ORNL-5870, Oak Ridge National Laboratory, Oak Ridge, TN, November.

Braun, J. B., S. J. Miller and B. L. Ringe, 1993, *Historically Significant Scientific and Technical Facilities at the INEL (Draft)*, EGG-CS-10699, EG&G Idaho, Inc., Idaho Falls, ID, March.

Breslin, S., 1993, Texas Natural Heritage Program, Texas Parks and Wildlife Department, personal communication to M. Shafer, Science Applications International Corporation, September 15.

Butz, T. W., 1984, *Resource Management Plan for DOE Oak Ridge Reservation, Volume 8, Appendix H: Geology*, ORNL-6026/V8, Martin Marietta Energy Systems, Inc., Oak Ridge, TN, July.

Cartwright, P. (United Kingdom Atomic Energy Authority), 1995, personal communication to H. Chernoff, Science Applications International Corporation, March 16.

Cartwright, P. (United Kingdom Atomic Energy Authority), 1994, personal communication to A. Murray, Science Applications International Corporation, November 8.

Cholewa, A. F. and D. M. Henderson, 1984, *A Survey and Assessment of the Rare Vascular Plants of the INEL*, DOE/ID-12100, U.S. Department of Energy, Idaho Operations Office, Radiological and Environmental Sciences Laboratory, Idaho Falls, ID, July.

Clawson, K. L., G. E. Start and N. R. Ricks, 1989, *Climatography of the INEL*, second edition, DOE/ID-12118, National Oceanic and Atmospheric Administration, Idaho Falls, ID, December.

COGEMA, 1994, *Nuclear Materials Reprocessing and Recycling*, Paris, France, November.

Cottam, R., 1995, Lockheed Idaho Technology Company, personal communication to P. Schwab, Science Applications International Corporation, Germantown, MD, January 5.

CRC (CRC Press), 1991, *CRC Handbook of Chemistry and Physics*, 72nd Edition, 1991-1992, Boca Raton, FL.

Cummins, C. L., D. K. Martin and J. L. Todd, 1990, *Savannah River Site Environmental Report for 1989*, WSRC-IM-90-60, Westinghouse Savannah River Company, Aiken, SC.

Cunningham, M. and L. Pounds, 1991, *Resource Management Plan for the Oak Ridge Reservation, Volume 28, Wetlands on the Oak Ridge Reservation*, ORNL/NERP-5, Martin Marietta Energy Systems, Inc., Oak Ridge, TN, December.

Dahlke, H. J., D. A. Johnson, J. K. Rawlins, D. K. Searle, and G. W. Wachs, 1994, *Summary Engineering Description of Underwater Fuel Storage Facility for Foreign Research Reactor Spent Nuclear Fuel*, INEL-94/0014, Idaho Falls, ID, October.

Denney, R., 1995, Lockheed Idaho Technology Company, personal communication to A. Zeitoun, Science Applications International Corporation, Germantown, MD, January 25.

Department of the Navy, 1994, *Environmental Assessment, Construction of Magazines and Rail Lines, Naval Weapons Station, Charleston, South Carolina*, Public Works Officer, Naval Weapons Station, Charleston, SC, July.

Department of the Navy, 1990, *Environmental Assessment for Propulsion Training Facility, Naval Weapons Station Charleston, Berkeley County, South Carolina*, Southern Division, Naval Facility Engineering Command, Charleston, SC, July.

de Planque, E. G., 1994, *In Search of ... Background*, Remarks before the NRC Workshop on Site Characterization for Decommissioning, U.S. Nuclear Regulatory Commission, Rockville, MD, November 29.

DOC (U.S. Department of Commerce, Bureau of the Census), 1994, *Statistical Abstract of the United States, 114th Edition*, Washington, DC.

DOE (U.S. Department of Energy), 1995a, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE/EIS-0220, Savannah River Site, Aiken, SC, October.

DOE (U.S. Department of Energy), 1995b, *Draft Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste*, DOE/EIS-0200-D, Office of Environmental Management, Washington, DC, August.

DOE (U.S. Department of Energy), 1995c, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*, DOE/EIS-0203-F, Idaho Operations Office, Idaho Falls, ID, April.

DOE (U.S. Department of Energy), 1994a, *Integrated Data Base Report-1993: U.S. Spent Nuclear Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, DOE/RW-0006, Rev. 10, Office of Civilian Radioactive Waste Management and Office of Environmental Management, Washington, DC, December.

DOE (U.S. Department of Energy), 1994b, *Final F-Canyon Plutonium Solutions Environmental Impact Statement*, DOE/EIS-0219, Savannah River Site, Aiken, SC, December.

DOE (U.S. Department of Energy), 1994c, *DOE Spent Nuclear Fuel Technology Integration Plan*, SNF-PP-FS-002, Office of Environmental Management, Washington, DC, December.

DOE (U.S. Department of Energy), 1994d, *Shipment Mobility and Accountability Collection (SMAC) Special Report*, Oak Ridge, TN, December 1.

DOE (U.S. Department of Energy), 1994e, *Final Supplemental Environmental Impact Statement, Defense Waste Processing Facility*, DOE/EIS-0082-S, Savannah River Site, Aiken, SC, November.

DOE (U.S. Department of Energy), 1994f, *A Disposition Strategy for Highly Enriched Aluminum-Based Fuel from Research and Test Reactors*, Savannah River Site, Aiken, SC, November 10.

DOE (U.S. Department of Energy), 1994g, *Financial Assistance to States and Tribes to Support Emergency Preparedness and Response and the Safe Transportation of Hazardous Shipments*, TTP No. RL-327001, Richland, WA, October.

DOE (U.S. Department of Energy), 1994h, *Implementation Plan for the Environmental Impact Statement on the Proposed Policy for the Acceptance of Foreign Research Reactor Spent Nuclear Fuel*, DOE/EIS-0218, Washington, D.C., October.

DOE (U.S. Department of Energy), 1994i, *Transportation External Coordination/Working Group (TEC/WG) List of Invited Organizations and Previous Participants*, Germantown, MD, September 28.

DOE (U.S. Department of Energy), 1994j, *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Draft Environmental Impact Statement*, DOE/EIS-0203-D, Idaho Operations Office, Idaho Falls, ID, June.

DOE (U.S. Department of Energy), 1994k, *Safety Analysis Report (200-S Area), Savannah River Site Defense Waste Processing Facility Operations*, DPSTSA-200-10, Supplement 20, Savannah River Site, Aiken, SC, June.

DOE (U.S. Department of Energy), 1994l, "Notice of Intent to Prepare a Programmatic Environmental Impact Statement for Storage and Disposition of Weapons-Usable Fissile Materials," *Federal Register*, Volume 59, No. 118, June 21.

DOE (U.S. Department of Energy), 1994m, *Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel*, DOE/EA-0912, Washington, DC, April.

DOE (U.S. Department of Energy), 1994n, *Report on the Emergency Response Training and Equipment Activities through Fiscal Year 1993 for the Transportation of Transuranic Waste to the Waste Isolation Pilot Plant*, DOE/WIPP 93-061, Carlsbad, NM, April.

DOE (U.S. Department of Energy), 1994o, *Waste Acceptance System Requirements Document*, Revision 01, DOE/RW-0351P Rev. 1, February.

DOE (U.S. Department of Energy), 1993a, *Annual Site Environmental Report - 1992, Volume 1 and Volume 2*, DOE/NV/10630-66, Nevada Operations Office, Las Vegas, NV, September.

DOE (U.S. Department of Energy), 1993b, *1993 NTS Technical Site Information Package - Volumes 1 and 2*, Nevada Operations Office, Las Vegas, NV, September.

DOE (U.S. Department of Energy), 1993c, *Tornado Special Study*, ES/ESH-35, June 14.

DOE (U.S. Department of Energy), 1993d, *Nonnuclear Consolidation Environmental Assessment, Volume 1, Nuclear Weapons Complex Reconfiguration Program*, DOE/EA-0792, Office of Defense Programs, Washington, DC, June.

DOE (U.S. Department of Energy), 1993e, *Transporting Radioactive Materials - Q&A to Your Questions*, DOE\EM-0097, Washington, DC, April.

DOE (U.S. Department of Energy), 1992a, *DOE Standard: Hazard Categorization Accident Analysis Technique for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Report*, DOE-STD-1027-92, December.

DOE (U.S. Department of Energy), 1992b, *Integrated Data Base for 1992: U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, DOE/RW-006, Rev. 8, Office of Civilian Radioactive Waste Management, Washington, DC, October.

DOE (U.S. Department of Energy), 1992c, *U.S. Department of Energy Nevada Field Office Annual Site Environmental Report - 1991*, DOE/NV/10630-33, Nevada Operations Office, Las Vegas, NV, September.

DOE (U.S. Department of Energy), 1991a, *Draft Environmental Impact Statement for the Siting, Construction and Operation of New Production Reactor Capacity*, DOE/EIS-0144D, Office of New Production Reactors, Washington, DC, April.

DOE (U.S. Department of Energy), 1991b, *U.S. Proposal for the Nuclear Weapons Complex Reconfiguration Site*, Savannah River Field Office, Aiken, SC.

DOE (U.S. Department of Energy), 1991c, *Planning and Preparedness for Operational Emergencies*, DOE Order 5500.3A, April 30.

DOE (U.S. Department of Energy), 1990a, *Final Environmental Impact Statement, Continued Operation of K-, L-, and P-, Reactors, Savannah River Plant, Aiken, South Carolina*, DOE/EIS-0147, Savannah River Operations Office, Aiken, SC, December.

DOE (U.S. Department of Energy), 1990b, *Review of the Emergency Management System and Its Preparedness to Respond to Transportation Incidents,* TITF-91-001, November.

DOE (U.S. Department of Energy), 1990c, *Radiation Protection of the Public and the Environment*, DOE Order 5400.5, February 8.

DOE (U.S. Department of Energy), 1989, *Shipment of Radioactive Material by the U.S. Department of Energy,* DOE/DP-0065, Washington, DC.

DOE (U.S. Department of Energy), 1988a, *Site Characterization Plan, Yucca Mountain Site, Nevada Research and Development Area, Nevada*, DOE/RW-0199, Office of Civilian Radioactive Waste Management, December.

DOE (U.S. Department of Energy), 1988b, *Status of Flora and Fauna on the Nevada Test Site*, DOE/NV/10630-29, Nevada Operations Office, Las Vegas, NV.

DOE (U.S. Department of Energy), 1988c, *Consultant Draft: Site Characterization Plan, Reference Repository Location, Hanford Site, Washington*, DOE/RW-0164, Washington, DC, January.

DOE (U.S. Department of Energy), 1987, *Final Environmental Impact Statement, Waste Management Activities for Groundwater Protection, Savannah River Plant, Aiken, South Carolina*, DOE/EIS-0120, Savannah River Operations Office, Aiken, SC, December.

DOE (U.S. Department of Energy), 1986a, *Environmental Assessment, Reference Repository Location, Hanford Site, Washington,* DOE-RW-0070, Office of Civilian Radioactive Waste Management, Washington, DC, May.

DOE (U.S. Department of Energy), 1986b, *Environmental Assessment, Yucca Mountain Site, Nevada Research and Development Area, Nevada*, DOE/RW-0073, Office of Civilian Radioactive Waste Management, Washington, DC, May.

DOE (U.S. Department of Energy), 1982, *Geology of the Nevada Test Site and Nearby Areas, Southern Nevada*, SAND82-2207, Sandia National Laboratories, October.

DOE (U.S. Department of Energy), 1980, *Final Environmental Impact Statement, United States Spent Fuel Policy*, DOE/EIS-0015, Washington, DC, May.

Dukes, E. K., 1984, *The Savannah River Plant Environment*, DP-1642, E.I. du Pont de Nemours and Company, Inc., Aiken, SC, March.

Dupont, M.E., 1996, Westinghouse Savannah River Company, personal communication to I. Zeitoun, Science Applications Internatinal Corporation, January 10.

Eckel, E. B., 1968, *The Geologic Society of America, Inc. Memoir 110, Nevada Test Site*, The Geologic Society of America, Boulder, CO.

EG&G (Edgerton, Germeshausen, and Grier) Idaho, Inc., 1993, *Waste Management Facilities Cast Information Estimating Data for Spent Nuclear Fuel*, EGG-WM-10708, March.

EG&G (Edgerton, Germeshausen, and Grier) Idaho, Inc., 1984, *INEL Environmental Characterization Report*, EGG-NPR-6688, Idaho Falls, ID, September.

EPA (U.S. Environmental Protection Agency), 1993a, *Attainment Status Designations, South Carolina*, 40 CFR 81(Revised as of July 1, 1992), Office of the Federal Register, Washington, DC.

EPA (U.S. Environmental Protection Agency), 1993b, *Prevention of Significant Deterioration of Air Quality*, 40 CFR 52.21 (Revised as of July 1, 1992), Office of the Federal Register, Washington, DC.

EPA (U.S. Environmental Protection Agency), 1991a, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, EPA 400-R-92-001, Office of Radiation Programs, Washington, DC, October (Reprinted May 1992).

EPA (U.S. Environmental Protection Agency), 1991b, "Sole Source Designation of the Eastern Snake River Plain Aquifer, Southern Idaho, Final Determination," *Federal Register*, Volume 56, No. 194, October.

ERDA (U.S. Energy Research and Development Administration), 1976, *Ecology of the Nevada Test Site: A Narrative Summary and Annotated Bibliography*, Las Vegas, NV, May.

FEMA (Federal Emergency Management Agency), 1994, *Cooperative Contract Agreement Funding Documents for Fiscal Years 1982-1993,* October.

FEMA (Federal Emergency Management Agency), 1985, "Federal Radiological Emergency Response Plan," *Federal Register*, Volume 50, No. 217, November 8.

FHI (Frederick R. Harris, Inc.), 1993, *Kings Bay Naval Station Submarine Base; Assessment for Receipt of FRR SNF*, Iselin, NJ, November 1.

FICON (Federal Interagency Committee on Noise), 1992, *Federal Agency Review of Selected Airport Noise Analysis Issues*, U.S. Department of Transportation, Federal Aviation Administration, Washington, DC.

Fitzner, R. E. and R. H. Gray, 1991, "The Status, Distribution, and Ecology of Wildlife on the U.S. DOE Hanford Site: A Historical Overview of Research Activities," *Environmental Monitoring and Assessment*, Volume 18, pp. 173-202.

Friedrich, H., 1969, *Marine Biology: An Introduction to Its Problems and Results*, University of Washington Press, Seattle, WA.

FWS (U.S. Fish and Wildlife Service), 1992, *Threatened and Endangered Species of Texas*, Texas State Office, Austin, TX, August.

FWS (U.S. Fish and Wildlife Service), 1981a, *Pacific Coast Ecological Inventory Map*: 38120-A1-EI-250, Sacramento, CA.

FWS (U.S. Fish and Wildlife Service), 1981b, *Pacific Coast Ecological Inventory Map*: 38122-A1-EI-250, Santa Rosa, CA.

FWS (U.S. Fish and Wildlife Service), 1981c, Pacific Coast Ecological Inventory, 45122-A1-E1-250, Vancouver, Washington, Oregon.

FWS (U.S. Fish and Wildlife Service), 1980a, *James Island Atlantic Coast Ecological Inventory*, 32078-A1-EI-250, James Island, SC.

FWS (U.S. Fish and Wildlife Service), 1980b, *Atlantic Coast Ecological Inventory*, 30080-A1-EI-250, Jacksonville, FL-GA.

FWS (U.S. Fish and Wildlife Service), 1980c, *Savannah, Atlantic Coast Ecological Inventory Map*, 32080-A1-EI-250, Savannah, GA.

Gaia Northwest, Inc., 1988, *Drinking Water Consumption and Alternative Sources for the Eastern Snake River Plain, Idaho*, November.

Gaines, E., 1994, Oregon National Heritage Program, personal communication to D. Scott, Science Applications International Corporation, April 12.

GAO (U.S. General Accounting Office), 1994, *Foreign Countries' Approaches to High-Level Waste Storage and Disposal*, GAO/RCED-94-172, Washington, DC, August.

Garabedian, S. P., 1989, *Hydrology and Digital Simulation of the Regional Aquifer System, Eastern Snake River Plain, Idaho*, USGS Open File Report 87-237, U.S. Geological Survey.

Garabedian, S. P., 1986, *Application of a Parameter Estimation Technique to Modeling the Regional Aquifer Underlying the Eastern Snake River Plain, Idaho*, Water Supply Paper 2278, U.S. Geological Survey.

Gilbert, H. K. and B. L. Ringe, 1993, *Inventory of Known Historical Cultural Resources on the INEL and Preliminary Analysis of Historic Sensitivity (Draft)*, EGG-CS-10707, EG&G Idaho Inc., Idaho Falls, ID, March.

Golder Associates, 1993, *Assessment of Trends in Groundwater Quality at the Idaho National Engineering Laboratory*, Report No. 933-1151, Idaho Falls, ID, October 29.

Gonzalez, A., J. Pierce and D. Stenberg, 1986, *Target Hardness Comparison to IAEA Unyielding Target*, IAEA-SM-114, SAND86-0052c, Sandia National Laboratories, Albuquerque, NM.

Gordon-Hagerty, L. E., 1993, "United States Department of Energy Radiological Emergency Response Programme - A National Capability," *Radiation Protection Dosimetry*, Volume 50, Nos. 2-4, pp. 153-156.

Graham, M. J., M. D. Hall, S. R. Strait and W. R. Brown, 1981, *Hydrology of the Separations Area*, RHO-ST-42, Rockwell Hanford Operations, Richland, WA, July.

Grainger, A. R., 1995, *Radiological Emissions Calculations for Savannah River Site (SRS) Processing Activities*, Memorandum to Charles Head from Savannah River Operations Office, July 31, 1995.

Greene, G. S., 1975, *Prehistoric Utilization of the Channeled Scablands of Eastern Washington*, Ph.D Dissertation, Washington State University, Department of Anthropology, Pullman, WA.

Greger, P. D., E. M. Romney, n.d.a., *Wildlife Utilization of National Springs and Man-Made Sources at the Nevada Test Site (Draft)*, UCLA Laboratory of Biomedical and Environmental Sciences.

Hackett, W. R. and R. P. Smith, 1992, "Quaternary Volcanism, Tectonics, and Sedimentation in the Idaho National Engineering Laboratory Area," in *Field Guide to Geologic Excursions in Utah and Adjacent Areas of Nevada, Idaho, Wyoming*, by J. R. Wilson, Geological Society of America, Rocky Mountain Section Guidebook, Utah Geological Survey Miscellaneous Publication 92-3, pp. 1-18.

Hajek, B. F., 1966, *Soil Survey: Hanford Project in Benton County, Washington*, BNWL-243, Pacific Northwest Laboratory, Richland, WA, April.

Hatcher, Jr., R. D., P. J. Lemiszki, R. B. Dreier, R. H. Ketelle, R. R. Lee, D. A. Lietzke, W. M. McMaster, J. L. Foreman and S. Y. Lee, 1992, *Status Report on the Geology of the Oak Ridge Reservation*, Oak Ridge National Laboratory, Environmental Services Division Publication No. 3860, ORNL/TM-12074, October.

Hessler, R. R. and H. L. Sanders, 1967, "Faunal Diversity in the Deep Sea," *Deep Sea Resources*, Volume 14, pp. 65-78.

HNUS (Haliburton NUS Environmental Corporation), 1993, *Forecasts Generated From the Regional Economic Models, Economic-Demographic Forecasting and Simulation Model*, Savannah River Center, Aiken, SC.

HNUS (Haliburton NUS Environmental Corporation), 1990, *Sound Level Characterization of the Savannah River Site*, NUS-5251, Savannah River Center, Aiken, SC, August.

Hoff, D. L., R. G. Mitchell, G. C. Bowman and R. Moore, 1990, *The Idaho National Engineering Laboratory Site Report for Calendar Year 1989*, DOE/ID-12082(89), Idaho Operations Office, June.

Holm, J. A., 1994, "Transportation External Coordination/Working Group," U.S. Department of Energy Transportation Management Workshop, Rockville, MD, August 9.

IAEA (International Atomic Energy Agency), 1992, *Effects of Ionizing Radiation on Plants and Animals at Levels Implied by Current Radiation Protection Standards*, Technical Report Series No. 332, Vienna, Austria.

IAEA (International Atomic Energy Agency), 1990, *Regulations for the Safe Transport of Radioactive Material*, 1985 Edition as Amended 1990, Safety Series No. 6, Vienna, Austria.

IAEA (International Atomic Energy Agency), 1988, *Assessing the Impact of Deep Sea Disposal of Low Level Radioactive Waste on Living Marine Resources*, Technical Reports Series No. 288, Vienna, Austria.

IAEA (International Atomic Energy Agency), 1976, *Effects of Ionizing Radiation on Aquatic Organizations and Ecosystems*, Technical Report Series No. 172, Vienna, Austria.

ICRP (International Commission on Radiological Protection), 1991, *1990 Recommendations of the International Commission on Radiological Protection*, Annals of the ICRP, Publication 60, Volume 21, No. 1-3, Pergamon Press, New York, NY, November.

IMO (International Maritime Organization), 1994, *International Maritime Dangerous Goods Code*, London, UK.

IMO (International Maritime Organization), 1993, Resolution A, 748 (18), *Code for the Safe Carriage of Irradiated Nuclear Fuel, Plutonium, and High-Level Radioactive Wastes in Casks on Board Ships*, London, UK, adopted November 4.

IMO (International Maritime Organization), 1992, *SOLAS - Consolidated Text of the International Convention for the Safety of Life at Sea*, London, UK.

IPA (Information Please Almanac), 1993, *The 1993 Information Please Almanac, Atlas, and Yearbook*, 46th Edition, Houghton Mifflin Company, Boston, MA.

Irving, J. S., 1993, *Environmental Resource Document for the Idaho National Engineering Laboratory*, Volume 1, EGG-WMO-10279, July.

Jane's (Jane's Information Group, Inc.), 1992, *Jane's Containerization Directory*, 24th Edition, 1992-1993, Alexandria, VA.

Jefferson, R. M. and H. R. Yoshimura, 1978, *Crash Testing of Nuclear Fuel Shipping Containers*, SAND77-1462, Sandia National Laboratories, Albuquerque, NM, February.

Johnson, P. E., D. S. Joy, D. B. Clark and J. M. Jacobi, 1993a, *HIGHWAY 3.1, An Enhanced Highway Routing Model: Program Description, Methodology, and Revised User's Manual*, ORNL/TM-12124, Oak Ridge National Laboratory, Oak Ridge, TN, March.

Johnson, P. E., D. S. Joy, D. B. Clark and J. M. Jacobi, 1993b, *INTERLINE 5.0, An Expanded Railroad Routing Model: Program Description, Methodology, and Revised User's Manual*, ORNL/TM-12090, Oak Ridge National Laboratory, Oak Ridge, TN, March.

Jones, G. P., R. W. Allott, V. J. Dennis, J. P. Fowler, B. Hughes, D. Lomas and J. Wilkinson, 1994, *Radioactive waste discharges from UKAEA establishments during 1993 and associated monitoring results*, UKAEA/CSD R012, United Kingdom Atomic Energy Authority, August.

Jow, H. N., J. L. Sprung, J. A. Rollstin, L. T. Ritchie and D. I. Chanin, 1990, *MELCOR Accident Consequences Code System (MACCS), Model Description*, NUREG/CR-4691, SAND86-1562, Volume 2, Sandia National Laboratories, Albuquerque, NM, February.

Kennett, J. P., 1982, *Marine Geology*, Prentice-Hall, Englewood Cliffs, NJ.

Ketelle, R. H. and D. D. Huff, 1984, *Site Characterization of the West Chestnut Ridge Site*, ORNL/TM-9229, Oak Ridge National Laboratory, Oak Ridge, TN.

King, A. L., D. J. Awl and C. A. Gabrielsen, 1994, *Environmentally Sensitive Areas Survey Program Threatened and Endangered Species Status Report*, Health Resources Division, Oak Ridge National Laboratory, Environmental Restoration Program (ES/ER/TM-130), September.

Knox, J. N. and R. R. Sharitz, 1990, *Endangered, Threatened, and Rare Vascular Flora of the Savannah River Site*, Savannah River Site National Environmental Research Park Publication, SRO-NERP-20, Aiken, SC.

Koch, H. F., 1984, *"Sinkholes: Their Geology, Engineering and Environmental Impact,"* Proceedings of First Multidisciplinary Conference on Sinkholes, Orlando, FL, October.

Kroodsma, R. L., 1987, *Resource Management Plan for the Oak Ridge Reservation: Volume 24, Threatened and Endangered Animal Species*, ORNL/ESH-1/V24, Oak Ridge National Laboratory, Oak Ridge, TN, January.

Kuntz, M. A., 1992, "A Model-based Perspective of Basaltic Volcanism, Eastern Snake River Plain, Idaho," in *Regional Geology of Eastern Idaho and Western Wyoming*, by P. K. Link, M. A. Kuntz and L. B. Platt, editors, Geological Society of America, Memoir 179, pp. 289-304.

Kurkoski, D., 1994, U.S. Army Corps of Engineers, personal communication to D. Scott, Science Applications International Corporation, March 24.

Lee, R. R. and R. H. Ketelle, 1987, *Stratigraphic Influence on Deep Groundwater Flow in the Knox Group Copper Ridge Dolomite on the West Chestnut Ridge Site*, ORNL/TM-10479, Oak Ridge National Laboratory, Oak Ridge, TN, October.

Leenheer, J. A. and J. C. Bagby, 1982, *Organic Solutes in Groundwater at the Idaho Engineering Laboratory*, Water Resources Investigation Report 82-15, U.S. Geological Survey, ID022061, Idaho Falls, ID, March.

Leonard, P. R., 1993, *Estimated Radiological Doses Resulting from Airborne Radionuclides Released by Facilities at the INEL*, EGG-WTD-100676, EG&G Idaho, Inc., Idaho Falls, ID, July.

Leventhal, P. L. and E. S. Lyman, 1995, Nuclear Control Institute, personal communication to H. O'Leary, U.S. Department of Energy, December 20.

Lewis, B., 1995, Naval Weapons Station Charleston, personal communication to G. DeMoss, Science Applications International Corporation, October 26.

Liszewski, M. J. and L. J. Mann, 1992, *Purgeable Organic Compounds in Groundwater at the Idaho National Engineering Laboratory*, DOE/ID 22104, Idaho Falls, ID, July.

Lloyd's (Lloyd's of London Maritime Information and Consultancy Services), 1991, *1991 Casualty File Manual,* London, England, November 18.

Mann, L. J. and L. L. Knobel, 1987, *Purgeable Organic Compounds at the Idaho National Engineering Laboratory*, Open File Report 766, U.S. Geological Survey, DOE/ID 22074, Idaho Falls, ID, December.

Masson, H., S. Runge, S. E. Ion and N. L. Griffin, 1994, "Waste Management Issues: Reprocessing or Direct Disposal," *Volume 1, RECOD '94*, Proceedings of Fourth International Conference on Nuclear Fuel Reprocessing and Waste Management, April.

Matos, J. E., 1994, *Foreign Research Reactor Irradiated Nuclear Fuel Inventories Containing HEU and LEU of U.S. Origin*, Argonne National Laboratory, ANL/RERTR/TM-22, December.

McBee, A., 1994, South Carolina Heritage Trust, South Carolina Wildlife & Marine Resources Department, personal communication to D. Scott, Science Applications International Corporation, April 5.

McKibben, J. M., T. H. Gould, W. R. McDonell, and W. E. Bickford, 1994, *A Disposition Strategy for Highly Enriched Aluminum-Based Fuel from Research and Test Reactors,* Westinghouse Savannah River Company, Aiken, SC, November.

McKinney, J. D., 1985, *Big Lost River 1983-1984 Flood Threat,* PPD-FPB-002, EG&G, Idaho Falls, ID, July.

MMES (Martin Marietta Energy Systems), 1994, *White Paper Summary Report: Spent Nuclear Fuels Management Complex Siting Study*, Oak Ridge, TN, March.

MMES (Martin Marietta Energy Systems), 1993a, *Oak Ridge Reservation Technical Site Information Draft,* ES/EN/SFP-23, Site and Facilities Planning Department, Oak Ridge, TN, September.

MMES (Martin Marietta Energy Systems), 1993b, *Oak Ridge Reservation Environmental Report for 1992,* ES/ESH-31, Oak Ridge, TN, June.

MMES (Martin Marietta Energy Systems), 1992, Oak Ridge Reservation Environmental Report for 1991, ES/ESH-22, October.

MMES (Martin Marietta Energy Systems), 1991, *Oak Ridge Reservation Site Development and Facilities Utilization Plan, 1990 Update,* DOE/OR-885(91), Oak Ridge, TN, June.

MMES (Martin Marietta Energy Systems), 1989, *Oak Ridge Reservation Site Development and Facilities Utilization Plan,* DOE/OR-885/R1, Oak Ridge, TN, June.

Murray, S., 1994, Florida Natural Areas Inventory, personal communication to D. Scott, Science Applications International Corporation, April 27.

NCRP (National Council on Radiation Protection and Measurements), 1991, *Effects of Ionizing Radiation on Aquatic Organisms,* NCRP Report No. 109, Bethesda, MD, March.

NEA (Nuclear Energy Agency), 1988, *Feasibility of Disposal of High-Level Radioactive Waste into the Seabed,* Office of Economic Cooperation and Development.

Neuhauser, K. S. and F. L. Kanipe, 1993, *RADTRAN 4, Volume II: Technical Manual*, SAND 89-2370, Sandia National Laboratories, Albuquerque, NM, August.

NNPP (Naval Nuclear Propulsion Program), 1993, *Occupational Radiation Exposure for U.S. Naval Nuclear Plants and Their Support Facilities*, Report NT-93-2, Washington, DC, February.

NOAA (National Oceanic and Atmospheric Administration), 1993a, *Local Climatological Data: Annual Summary with Comparative Data, Galveston, Texas*, National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1993b, *Local Climatological Data: Annual Summary with Comparative Data, Los Angeles, California*, National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992a, *Local Climatological Data Annual Summary with Comparative Data - Charleston, South Carolina*, National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992b, *Local Climatological Data Annual Summary with Comparative Data - Jacksonville, Florida,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992c, *Local Climatological Data: Annual Summary with Comparative Data, Norfolk, Virginia,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992d, *Local Climatological Data Annual Summary with Comparative Data - Portland, Oregon,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992e, *Local Climatological Data Annual Summary with Comparative Data - Savannah, Georgia,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992f, *Local Climatological Data Annual Summary with Comparative Data - Seattle, Washington,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992g, *Local Climatological Data Annual Summary with Comparative Data - Wilmington, North Carolina*, National Climatic Data Center, Asheville, NC.

NRC (U.S. Nuclear Regulatory Commission), 1995, *Nuclear Regulatory Commission, 10 CFR Part 71, Compatibility with International Atomic Energy Agency (IAEA)*, Federal Register, Volume 60, No. 188, September 28.

NRC (U.S. Nuclear Regulatory Commission), 1993, *Public Information Circular for Shipments of Irradiated Nuclear Fuel*, NUREG-0725, Revision 9, Washington, DC, March.

NRC (U.S. Nuclear Regulatory Commission), 1991, *Standards for Protection Against Radiation* (Revised), 10 CFR Parts 2, 19, 20, 30, 31, 32, 34, 35, 39, 40, 50, 61, and 70, Washington, DC, December.

NRC (U.S. Nuclear Regulatory Commission), 1987, *Transporting Spent Fuel Protection Provided Against Severe Highway and Railroad Accidents*, NUREG/BR-0111, Washington, DC, March.

NRC (U.S. Nuclear Regulatory Commission), 1977, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes*, NUREG-0170, Volumes I and II, Office of Standards Development, Washington, DC, December.

NRC (U.S. Nuclear Regulatory Commission), 1975, *Spent Fuel Storage Facility Design Basis*, Regulatory Guide 1.13, Revision 1, December.

ORNL (Oak Ridge National Laboratory), 1988, *Data Package for the Low-Level Waste Disposal Development and Demonstration Program, Environmental Impact Statement*, ORNL/RM-10939, Volume 1 and Volume 2, Oak Ridge, TN, September.

Orr, B. R. and L. D. Cecil, 1991, *Hydrologic Conditions and Distribution of Selected Chemical Constituents in Water, Snake River Plain Aquifer, Idaho National Engineering Laboratory, 1986-1988,* Water Resource Report 91-4047, U.S. Geological Summary, Idaho Falls, ID, March.

Parr, P. D. and J. W. Evans, 1992, *Draft Resource Management Plan for the Oak Ridge Reservation, Wildlife Management Plan*, ORNL/NERP-6, Oak Ridge National Laboratory, Oak Ridge, TN.

Parr, P. D. and L. R. Pounds, 1987, *Resource Management Plan for the Oak Ridge Reservation*, ORNL/ESH-1/V23, Oak Ridge National Laboratory, Oak Ridge, TN.

PIC (Planning Information Corporation), 1992, *A Socioeconomic Assessment of the Proposed Caliente Rail Spur, Nye County (Draft)*, Nevada, December.

Pickard, G. L. and W. J. Emery, 1982 *Descriptive Physical Oceanography*, 4th Edition, Pergamon Press.

Pittman, J. R., R. G. Jensen and P. R. Fischer, 1988, *Hydrologic Conditions at the INEL, 1982-1985*, U.S. Geological Survey Water Resources Investigation Report 89-4008, DOE/ID-22078, December.

Pounds, L. R., P. D. Parr and M. G. Ryon, 1993, *Resource Management Plan for the Oak Ridge Reservation, Volume 30: Oak Ridge National Environmental Research Park Natural Areas and Reference Areas - Oak Ridge Reservation Environmentally Sensitive Sites Containing Special Plants, Animals, and Communities*, ORNL-NERP-8, Oak Ridge National Laboratory, Oak Ridge, TN, August.

POW (Port of Wilmington), 1994, Port of Wilmington, DE, Brochure and Fact Sheet.

REAC/TS (Radiation Emergency Assistance Center/Training Site), n.d.a., *The Radiation Emergency Assistance Center/Training Site and World Health Organization Collaborating Center for Radiation Emergency Assistance,* Oak Ridge, TN.

Reynolds, T. D., 1993, Environmental Science and Research Foundation, U.S. Department of Energy, Idaho Operations Office, personal communication with T. Doerr, Science Applications International Corporation, September 8.

Rice, A. L., 1978, "Radioactive Waste Disposal and Deep Sea Biology," *Oceanol*, Acta 1, pp. 483-491.

Rice, D. G., 1968, *Archaeological Reconnaissance: Ben Franklin Reservoir Area, 1968*, Washington State University, Laboratory of Anthropology, Pullman, WA.

Rickard, W. H. and D. G. Watson, 1985, "Four Decades of Environmental Change and Their Influences Upon Native Wildlife and Fish on the Mid-Columbia River, Washington, U.S.A.," *Environmental Conservation*, Volume 12, No. 3, pp. 241-248.

Robertson, J. B., R. Schoen and J. T. Barraclough, 1974, *The Influence of Liquid Waste Disposal on the Geochemistry of Water at the National Reactor Testing Station, Idaho: 1952-1970*, IDO-22053, U.S. Geological Survey, February.

Rogers, L. E. and W. H. Rickard, 1977, *Ecology of the 200 Area Plateau Waste Management Environs: A Status Report,* PNL-2253, Pacific Northwest Laboratory, Richland, WA, October.

Rogoff, C., 1994, U.S. Department of Transportation, Research & Special Projects Administration, personal communication to M. Zanotti, Science Applications International Corporation, Germantown, MD, October 12.

Rohay, A. C., 1987, *Earthquake Focal Mechanisms, Recurrence Rates and Deformation in the Columbia River Basalts*, RHO-BW-SA-666P, Rockwell Hanford Operations, Richland, WA.

Rosensteel, B., 1994, *Wetlands and Rare Plant Information*, Internal Correspondence, Oak Ridge National Laboratory, Oak Ridge, TN, March 21.

RSN (Raytheon Services Nevada), 1993, *Summary of Natural Resources that Potentially Influence Human Intrusion at the Area 5 Radioactive Waste Management Site, DOE/Nevada Test Site, Nye County, Nevada*, Las Vegas, NV, August.

Sackschewsky, M. R., D. S. Landeen, G. I. Baird, W. H. Rickard and J. L. Downs, 1992, *Vascular Plants of the Hanford Site*, WHC-EP-0554, Westinghouse Hanford Company, Richland, WA, July.

Schalles, J. F., R. R. Sharitz, J. W. Gibbons, G. J. Leversee and J. N. Knox, 1989, *Carolina Bays of the Savannah River Plant*, National Environmental Research Park Publications, Savannah River Ecology Laboratory, SRO-NERP-18, Aiken, SC.

Schmitt, D. N., 1993, *Savannah and Ogeechee River Creek Surveys*, Georgia Department of Natural Resources, Game and Fish Division.

Schultz, T. J., 1978, "Synthesis of Social Surveys on Noise Annoyance," *Journal of the Acoustical Society of America*, Volume 64, No. 2, pp. 377-405, August.

Scott, W. E., 1982, *Surficial Geologic Map of the Eastern Snake River Plain and Adjacent Areas, Idaho and Wyoming*, U.S. Geological Survey, Miscellaneous Investigation Map I-1372.

Shedrow, C. B., 1993, *SRS Affected Environment*, WSRC-TR-93-667, Westinghouse Savannah River Company, Aiken, SC, October.

Shields, J. D., N. D. Woody, A. S. Dicks, G. J. Hollod, J. Schalles and G. J. Leversee, 1982, *Locations and Areas of Ponds and Carolina Bays at the Savannah River Plant*, DP-1525 (Revision 1), E.I. du Pont de Nemours and Company, Inc., Savannah River Laboratory, Aiken, SC.

Shirai, E., M. Adachi, T. Kodaira and H. Sato, 1991, "Dry Storage Facility of JRR-3 Spent Fuels", Third International Conference on Nuclear Fuel Processing and Waste Management, *RECOD '91*, Sendai, Japan, April 14-18, 1991.

Silver, J. M., 1993, Australian Nuclear Science and Technology Organization, Facsimile to J. Edlow, Edlow International Company, April 8.

Slovic, P., 1986, "Informing and Educating the Public About Risk," *Risk Analysis*, Volume 6, No. 4, pp. 403-415.

Smith, R. B. and W. J Arabasz, 1991, "Seismicity of the Intermountain Seismic Belt", *in* Slemmons, D. B., E. R. Engdahl, M. D. Zoback, and D. D. Blackwell, eds., *Neotectonics of North America:* Boulder, CO, Geological Society of America, Decade Map, Volume 1.

Solomon, D. K., G. K. Moore, L. E. Toran, R. B. Drier and W. M. McMaster, 1992, *Status Report: A Hydrologic Framework for the Oak Ridge Reservation*, Oak Ridge National Laboratory, Environmental Services Division Publication No. 3815, ORNL/TM-12026, Oak Ridge, TN, May.

SSEB (Southern States Energy Board), 1994, *Transportation Working Group on the Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel Meeting Summary and Final Attendance List,* May 25-26.

State of North Carolina, 1994, Letter from R. M. Fry, Deputy Director of the Department of Environmental's Health and Natural Resources, to Ms. T. Vick, Blue Ridge Environmental Defense League, Glendale Springs, NC, October 10.

Stephenson, D. E., 1988, *August 1988 Savannah River Plant Earthquake*, DPST-88-841, E.I. du Pont de Nemours and Company, Inc., Savannah River Laboratory, Aiken, SC, September.

Stone, W. A., J. M. Thorp, O. P. Gifford and D. J. Hoitink, 1983, *Climatological Summary for the Hanford Area*, PNL-4622, Pacific Northwest Laboratory, Richland, WA, June.

Tallman, A. M., K. R. Fecht, M. C. Maratt and G. V. Last, 1979, *Geology of the Separation Areas Hanford Site, South-Central Washington*, RHO-ST-23, Rockwell Hanford Operations, Richland, WA, June.

Taylor, W., 1995, U.S. Department of Energy, personal communication to M. Zanotti, Science Applications International Corporation, Germantown, MD, January 20.

Taylor, W. and J. Hauptman, 1994, U.S. Department of Energy, personal communication to M. Zanotti, Science Applications International Corporation, Germantown, MD, September 28.

TDOT (Tennessee Department of Transportation), 1991, Tennessee City and County Maps Showing 1990 Average Daily Traffic, Bureau of Planning and Development, March 25.

TPWD (Texas Parks and Wildlife Department), 1989a, *Trends in Texas Commercial Fishery Landings,* 1977-1988, Management Data Series No. 7, Fisheries Division.

TPWD (Texas Parks and Wildlife Department), 1989b, *Trends in Finfish Landings by Sport-boat Fishermen in Texas Marine Waters,* May 1974-1988, Management Data Series No. 8, Fisheries Division.

U.S. Army (U.S. Army Corps of Engineers), 1991, *Environmental Impact Study, Savannah Harbor, Georgia Comprehensive Study*, July.

U.S. Federal Register, 1979, *Executive Order 12114 - Environmental Effects Abroad of Major Federal Actions*, 44 FR 1957, 3 CFR, 1979 Comp. p. 356, Washington, DC, January 4.

UBC (Uniform Building Code), *Uniform Building Code*, 1991, International Conference of Building Officials, Whittier, CA.

UKAEA (United Kingdom Atomic Energy Authority), 1994, *Capability of Dounreay Fuel Plants*, Government Division, Nuclear Sites Operations, Scotland, May.

USAF (U.S. Air Force), U.S. Navy and U.S. Department of Interior, 1991, *Special Nevada Report*, DE-AC08-88NV10715, Nellis Air Force Base, NV, September.

USDA (U.S. Department of Agriculture), 1992, *Savannah River Site, Wildlife, Fisheries, and Botany Operations Plan*, Savannah River Forest Station, New Ellenton, SC, December.

USMMA (U.S. Merchant Marine Academy) 1994, *Draft Report on the Workshop on Port Selection Criteria for Shipments of Foreign Research Reactor Spent Nuclear Fuel, Kings Point, NY,* Sandia National Laboratories, Albuquerque, NM, February 28.

WDW (Washington Department of Wildlife), 1994, *Priority Habitat and Species and Map*, Seattle South, No. 4712253, April.

Werner, F. T., 1994, U.S. Fish and Wildlife Service, Division of Ecological Services, personal communication to D. Scott, Science Applications International Corporation, November 2.

Wichmann, T., 1995, U.S. Department of Energy, Idaho Operations Office, personal communication to U.S. Department of Energy and contractors regarding Spent Nuclear Fuel Inventory Data, February 1.

Wilmot, E. L., J. D. McClure and R. E. Luna, 1981, *Report on a Workshop on Transportation Accident Scenarios Involving Spent Fuel*, SAND80-2012, TTC-0151, Transportation Analysis and Information Division, Sandia National Laboratories, Albuquerque, NM, February.

Wood, W. W. and W. H. Low, 1988, *Solute Geochemistry of the Snake River Plain Regional Aquifer System, Idaho and Eastern Oregon*, Professional Paper 1408-D, U.S. Geological Survey.

World Bank, 1994, *World Development Report, Infrastructure for Development*, Oxford University Press, New York, NY, June.

WSRC (Westinghouse Savannah River Company), 1995, *High-Level Waste System Plan Revision 5 (U)*, HLW-OVP-95-0031, Aiken, SC, April 26.

WSRC (Westinghouse Savannah River Company), 1994a, *Repository Emplacement Costs for Al-Clad HEU Spent Fuel (U)*, WSRC-TR-94-0230, Rev. 1, Aiken, SC, September 23.

WSRC (Westinghouse Savannah River Company), 1994b, *Life-Cycle Cost Analysis for Foreign Research Reactor Spent Nuclear Fuel Disposal (U)*, WSRC-TR-94-039, Aiken, SC, March 30.

WSRC (Westinghouse Savannah River Company), 1993a, *Technical Data Input, "Section 5.3.6, Geology,"* Draft 0, Rev. 0, Savannah River Site, Aiken, SC, October.

WSRC (Westinghouse Savannah River Company), 1993b, *Technical Data Input, "Section 5.3.8, Water Resources,"* Draft 0, Rev. 0, Savannah River Site, Aiken, SC, October.

Yocum, C., 1994, Concord Naval Weapons Station, CA, personal communication to R. L. Gotchy, Science Applications International Corporation, August 15 and 16.

Yohe, R., 1993, Idaho State Historical Preservation Office, personal communication to T. Rudolph, Science Applications International Corporation, Boise, ID.

Yuan, Y. C., S. Y. Chan, D. J. LePoire and R. Rothman, 1993, *RISKIND - A Computer Program for Calculating Radiological Consequences and Health Risks from Transportation of Spent Nuclear Fuel*, ANL/EAIS-06, Rev. 0, Argonne National Laboratory, Argonne, IL, February.

9. Glossary

Absorbed dose. The energy imparted by ionizing radiation per unit mass of irradiated material. The unit of absorbed dose is the rad.

Accident. An unplanned sequence of events that results in undesirable consequences.

Actinide. Any of a series of chemically similar, mostly synthetic, radioactive elements with atomic numbers ranging from actinium (89) through lawrencium (103).

Acute exposure. A single exposure to a toxic substance which may result in severe biological harm or death. Acute exposures are usually characterized as lasting no longer than a day.

Alpha-emitter. A radioactive substance that decays by releasing an alpha particle.

Alpha particle. A particle consisting of two protons and two neutrons, given off by the decay of many elements, including uranium, plutonium, and radon. Alpha particles cannot penetrate a sheet of paper. However, alpha emitting isotopes in the body can be very damaging.

As low as reasonably achievable (ALARA). The approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as is reasonable, taking into account social, technical, economic, practical, and public policy considerations. ALARA is not a dose limit but a process which has the objective of attaining doses as far below the applicable limits as is reasonably achievable.

Atomic Energy Act (AEA). A law passed in 1954 that placed nuclear production and control of nuclear materials within a civilian agency, originally the Atomic Energy Commission. The Atomic Energy Commission was replaced by the U.S. Nuclear Regulatory Commission, the U.S. Department of Energy, and predecessor agencies (i.e., ERDA, FERC).

Atomic number. The number of positively charged protons in the nucleus of an atom or the number of electrons on an electrically neutral atom.

Background radiation. Radiation from: (1) Naturally occurring radioactive materials which have not been technologically enhanced, (2) cosmic sources, (3) global fallout as it exists in the environment (such as from the testing of nuclear explosive devices), (4) radon and its progeny in concentrations or levels existing in buildings or the environment which have not been elevated as a result of current or prior activities, and (5) consumer products containing nominal amounts of radioactive material or producing nominal amounts of radiation.

Beta particle. A particle emitted in the radioactive decay of many radionuclides. A beta particle is identical with an electron. It has a short range in air and a low ability to penetrate other materials.

Canning. The process of placing spent nuclear fuel in canisters to retard corrosion, contain radioactive releases, or control geometry.

Cask. A heavily shielded massive container for holding nuclear materials during shipment.

Characterization. The determination of waste or spent nuclear fuel composition and properties, whether by review of process knowledge, nondestructive examination or assay, or sampling and analysis, generally done to determine appropriate storage, treatment, handling, transportation, and disposal requirements.

Chemical separation. A process for extracting uranium and plutonium from dissolved spent nuclear fuel and irradiated targets. The fission products that are left behind are high level wastes. Chemical separation is also known as reprocessing.

Cladding. The outer layer of metal over the fissile material of a nuclear fuel element. Cladding on the Department of Energy's spent fuel is usually aluminum, zirconium, or stainless steel.

Collective dose. The sum of the total effective dose equivalents of all individuals in a specified population. Collective dose is expressed in units of person-rem (or person-sievert).

Committed effective dose equivalent. The sum of the committed dose equivalents to various tissues in the body, each multiplied by the appropriate weighting factor. Committed effective dose equivalent is expressed in units of rem (or sievert), and will be accumulated during the fifty years following an intake of radioactive material into an individual's body.

Competitive fee. A fee that could be charged to foreign research reactor operators related to the estimated cost of spent nuclear fuel management and disposal outside the United States.

Conditioning. See stabilization (of spent nuclear fuel).

Contact-handled waste. Packaged waste whose external surface dose rate does not exceed 200 mrem per hour.

Contamination. The deposition of undesirable radioactive material on the surfaces of structures, areas, objects, or personnel.

Core. The central portion of a nuclear reactor containing the fuel elements, moderator, neutron poisons, and support structures.

Criticality. The conditions in which a system is capable of sustaining a nuclear chain reaction.

Cumulative impact. The impact on the environment which results from the incremental impact of the action when added to other past, present, and reasonably foreseeable future actions regardless of what agency or person undertakes such other actions. Cumulative impacts can result from individually minor but collectively significant actions taking place over a period of time.

Curie. The basic unit used to describe the intensity of radioactivity in a sample of material. The curie is equal to 37 billion disintegrations per second, which is approximately the rate of decay of 1 gram of the isotope radium-266. A curie is also a quantity of any radionuclide that decays at a rate of 37 billion disintegrations per second.

Decay (radioactive). Spontaneous disintegration of the nucleus of an unstable atom, resulting in the emission of particles and energy.

Decommissioning. Retirement of a nuclear facility, including decontamination and/or dismantlement.

Decontamination. Removal of unwanted radioactive or hazardous contamination by a chemical or mechanical process.

Degraded (spent nuclear fuel). See failed fuel.

Depleted uranium. Uranium that, through the process of enrichment, has been stripped of most of the uranium-235 it once contained, so that it has more uranium-238 than natural uranium. It is used as shielding, in some parts of nuclear weapons, and as a raw material for plutonium production.

Developed countries. Countries with high-income economies (World Bank, 1994).

Developing countries. Countries with other-than-high-income economies (World Bank, 1994).

Discounted dollars. Expressing income and expenditures that occur over time as if they occurred at a common point in time.

Disposal of fuel. Emplacement of fuel to ensure its isolation from the biosphere, with no intention of retrieval.

DOE Orders. Requirements internal to the U.S. Department of Energy (DOE) that establish DOE policy and procedures, including those for compliance with applicable laws.

Dose (or radiation dose). A generic term that means absorbed dose, dose equivalent, effective dose equivalent, committed effective dose equivalent, or total effective dose equivalent as defined elsewhere in this glossary.

Dose rate. The radiation dose delivered per unit time (e.g., rem per year).

Dry storage. Storage of spent nuclear fuel in environments where the fuel is not immersed in water for purposes of both cooling and shielding.

Ecology. The relationship of living things to one another and their environment, or the study of such relationships.

Effective dose equivalent. The summation of the products of the dose equivalent received by specified tissues of the body and the appropriate weighting factor. It includes the dose from radiation sources internal and/or external to the body. The effective dose equivalent is expressed in units of rem (or sievert).

Endangered species. Animals, birds, fish, plants, or other living organisms threatened with extinction by man-made or natural changes in their environment. Requirements for declaring a species endangered are contained in the Endangered Species Act.

Enriched uranium. Uranium that has greater amounts of the isotope uranium-235 than occurs naturally. Naturally occurring uranium is 0.72 percent uranium-235.

Environmental monitoring. The process of sampling and analysis of environmental media in and around a facility being monitored for the purpose of (1) confirming compliance with performance objectives and (2) early detection of any contamination entering the environment to facilitate timely remedial action.

Escalation. A real change in the price level of a particular good or service, unrelated to inflation.

Existing facilities. Facilities that existed at an active DOE site as of the Record of Decision for this Environmental Impact Statement.

Failed fuel. Spent nuclear fuel whose external cladding has cracked, pitted, corroded, or potentially allows the leakage of radioactive gases.

Fissile material. Any material fissionable by thermal (slow) neutrons; the two primary fissile isotopes are uranium-235 and plutonium-239.

Fission. The splitting or breaking of a nucleus into at least two other nuclei and the release of a relatively large amount of energy. Two or three neutrons are usually released during this type of transformation.

Fission products. The nuclei produced by fission of heavy elements, and their radioactive decay products.

Fissionable material. Commonly used as a synonym for fissile material, the meaning of this term has been extended to include material that can be fissioned by fast neutrons, such as uranium-238.

Fuel elements. Nuclear reactor fuel including both the fissile and the structural material serves as cladding.

Full-cost recovery fee. A fee that could be charged to foreign research reactor operators that recovers all costs incurred by the United States for management of their spent nuclear fuel.

Gamma ray. Very penetrating electromagnetic radiation of nuclear origin. Except for origin and energy level, identical to x-rays. Electromagnetic radiation frequently accompanying alpha and beta emissions as radioactive materials decay.

Geologic repository. A place to dispose of radioactive waste deep beneath the earth's surface.

Groundshine. The radiation dose received from radioactive material deposited on the ground's surface.

Half-life. The time in which one-half of the atoms of a particular radioactive substance disintegrate to another nuclear form.

Hazardous material. A substance or material in a quantity and form which may pose an unreasonable risk to health and safety or property when transported in commerce.

Hazardous substance. Any substance that when released to the environment in an uncontrolled or unpermitted fashion becomes subject to the reporting and possible response provisions of the Clean Water Act and the Comprehensive Environmental Response, Compensation, and Liability Act.

Hazardous waste. (1) Wastes that are identified or listed in 40 CFR 261.31 and 261.32. Source, special nuclear material, and by-product material as defined by the Atomic Energy Act of 1954, as amended, are specifically excluded from the term hazardous wastes. (2) As defined in RCRA, a solid waste, or combination of wastes, that because of its quantity, concentration, or physical, chemical, or infectious characteristics, may cause or significantly contribute to an increase in mortality or serious, irreversible, or incapacitating reversible illness or pose a substantial present or potential hazard to human health or the environment when improperly treated, stored, transported, or disposed of, or otherwise managed. (3) By-products of society that can pose a substantial or potential hazard to human health or the environment when improperly managed. Possesses at least one of four characteristics (ignitability, corrosivity, reactivity, or toxicity).

High-efficiency particulate air (HEPA) filter. A filter with an efficiency of at least 99.95 percent used to remove particles from air exhaust streams prior to releasing to the atmosphere.

Highly enriched uranium (HEU). Uranium with more than 20 percent of the uranium-235 isotope, used for making nuclear weapons and also as fuel for some isotope-production, research, naval propulsion, and power reactors.

High-level waste. The highly radioactive waste material that results from the reprocessing of spent nuclear fuel, including liquid waste produced directly from reprocessing and any solid waste derived from the liquid that contains a combination of transuranic and fission product nuclides in quantities that require permanent isolation. High-level waste may include the highly radioactive material that the NRC, consistent with existing law, determines by rule requires permanent isolation.

Inflation. A change in the nominal price level of all goods or services, unrelated to the real escalation of a particular good or service.

Isotopes. Different forms of the same chemical element that differ only by the number of neutrons in their nucleus. Most elements have more than one naturally occurring isotope. Many more isotopes have been produced in reactors and scientific laboratories.

Latent cancer fatalities (LCF). Deaths occurring at later years from radiation-induced cancers.

Levelization. Conversion of a stream of values that vary at a uniform rate over time to a constant value over the same period of time.

Life cycle costs. All costs except the cost of personnel occupying the facility incurred from the time that space requirement is defined until the facility passes out of the government's hands.

Low enriched uranium (LEU). Uranium enriched until it consists of up to 20 percent uranium-235. Used as nuclear reactor fuel.

Low-level waste. A catchall term for any radioactive waste that is not spent fuel, high-level, or transuranic waste.

Management (spent nuclear fuel). Emplacing, operating, and administering facilities, transportation systems, and procedures in order to ensure safe and environmentally responsible handling and storage of spent nuclear fuel pending (and in anticipation of a decision on ultimate disposition. Spent nuclear fuel management also includes activities such as stabilization, examination/characterization, processing or chemical separation, and research and development; including activities that may be necessary to prepare spent nuclear fuel for ultimate disposition.

Maximally exposed individual (MEI). A theoretical individual living at the site boundary receiving the maximum exposure. The individual is assumed to be located in a direction downwind from the release point.

Maximally exposed worker. A marine transport worker, port worker, ground transport worker, or onsite radiation worker who could receive the maximum radiation exposure in a given situation.

Maximum contaminant level (MCL). The maximum permissible levels of a contaminant in water which is delivered to the free flowing outlet of the ultimate user of a public water system, except in the case of turbidity where the maximum permissible level is measured at the point of entry to the distribution system. Contaminants added to the water under the circumstances controlled by the user, except those resulting from corrosion of piping and plumbing caused by water quality, are excluded from this definition.

Metric tons of heavy metal (MTHM). Quantities of unirradiated and spent nuclear fuel and targets are traditionally expressed in terms of metric tons of heavy metal (typically uranium), without the inclusion of other materials, such as cladding, alloy materials, and structural materials. A metric ton is 1,000 kilograms, which is equal to about 2,200 pounds.

National Environmental Policy Act. A Federal law, enacted in 1970, that requires the Federal government to consider the environmental impacts of, and alternatives to, major proposed actions in its decisionmaking processes. Commonly referred to by its acronym, NEPA.

Natural phenomena accidents. Accidents that are initiated by phenomena such as earthquakes, tornadoes, floods, etc.

Nearest public access individual (NPAI). A theoretical individual located at the point of nearest public access to a DOE facility, usually during an accident situation.

Net present value. The value of a series of future income and expense streams brought forward to the present at the discount rate.

Neutron. Uncharged elementary particles with a mass slightly greater than that of the proton, and found in the nucleus of every atom heavier than hydrogen.

Nonproliferation. Efforts to prevent or slow the spread of nuclear weapons and the materials and technologies used to produce them.

Normal operation. All normal conditions and those abnormal conditions that frequency estimation techniques indicate occur with a frequency greater than 0.1 events per year.

Nuclear fuel. Materials that are fissionable and can be used in nuclear reactors.

Plutonium. A manmade fissile element. Pure plutonium is a silvery metal that is heavier than lead. Material rich in the plutonium-239 isotope is preferred for manufacturing nuclear weapons, although any plutonium can be used. Plutonium-239 has a half-life of 24,000 years.

Population dose. See collective dose.

Probable maximum flood. The largest flood for which there is any reasonable expectancy in a specific area. The probable maximum flood is normally several times larger than the largest flood of record.

Processing (of spent nuclear fuel). Applying a chemical or physical process designed to alter the characteristics of the spent nuclear fuel matrix.

Public. Anyone outside the DOE site boundary at the time of an accident or during normal operation.

PUREX. An acronym for Plutonium-Uranium Extraction, the name of the chemical process usually used to reprocess spent nuclear fuel and irradiated targets.

Rad. The special unit of absorbed dose. One rad (0.01 gray) is equal to an absorbed dose of 100 ergs/gram.

Radiation (ionizing). Energy transferred through space or other media in the form of particles or waves. In this document, we refer to ionizing radiation which is capable of breaking up atoms or molecules. The splitting, or decay, of unstable atoms emits ionizing radiation.

Radioactive waste. Waste that is managed for its radioactive content; solid, liquid or gaseous material that contains radionuclides regulated under the AEA of 1954, as amended and of negligible economic value considering costs of recovery.

Radioactivity. The spontaneous emission of radiation from the nucleus of an atom. Radionuclides lose particles and energy through this process of radioactive decay.

Region of influence. Region in which the principal direct and indirect socioeconomic effects of actions are likely to occur and are expected to be of consequence for local jurisdictions.

Regulated substances. A general term used to refer to materials other than radionuclides that may be regulated by other applicable Federal, State, (or possibly local) requirements.

rem. Roentgen Equivalent Man which is a unit of dose equivalent. Dose equivalent in rem is numerically equal to the absorbed dose in rad multiplied by a quality factor, distribution factor and any other necessary modifying factor (1 rem = 0.01 sievert).

Reprocessing (spent nuclear fuel). See chemical separation.

Risk. Quantitative expression of possible loss that considers both the probability that a hazard causes harm and the consequences of that event.

Saltstone. Low-radioactivity fraction of high-level waste formed into a concrete block at the Savannah River Site.

Source material. (1) Uranium, thorium, or any other material that is determined by the Nuclear Regulatory Commission pursuant to the provisions of the Atomic Energy Act of 1954, Section 61, to be source material; or (2) ores containing one or more of the foregoing materials, in such concentration as the Nuclear Regulatory Commission may by regulation determine from time-to-time [Atomic Energy Act 11(z)].

Special nuclear material. (1) Plutonium, uranium enriched in the isotope 233 or in the isotope 235, and any other material that the Nuclear Regulatory Commission, pursuant to the provisions of the Atomic Energy Act of 1954, Section 51, determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing, but does not include source material.

Spent nuclear fuel. Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated.

Stabilization (of spent nuclear fuel). Actions taken to further confine or reduce the hazards associated with spent nuclear fuel, as necessary for safe management and environmentally responsible storage for extended periods of time. Activities which may be necessary to stabilize spent nuclear fuel include canning, processing, and passivation.

Storage. The collection and containment of waste or spent nuclear fuel in such a manner as not to constitute disposal of the waste or spent nuclear fuel for the purposes of awaiting treatment or disposal capacity (i.e., not short-term accumulation).

Surface water. All waters that are open to the atmosphere and subject to surface runoff. All waters naturally open to the atmosphere (rivers, lakes, reservoirs, streams, impoundments, seas, estuaries, etc.) and all springs, wells, or other collectors that are directly influenced by surface water.

Target. A tube, rod, or other form containing material that, on being irradiated in a nuclear reactor would produce a designed end product (i.e., uranium-238 produces plutonium-239 and neptunium-237 produces plutonium-238).

Target material. Residual material that is left after a target has been irradiated and dissolved, and the end product has been removed. In this EIS, target material contains enriched uranium and fission products.

Total effective dose equivalent. The sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Type B packaging. Packaging for radioactive material which meets the standards for Type A packaging and, in addition, meets the standards for the hypothetical accident conditions of transport as prescribed in 49 Code of Federal Regulations Part 173.398(c). This includes spent fuel casks.

Ultimate disposition. The final step in which a material is either processed for some use or disposed of.

Undiscounted dollars. Expressing income and expenditures in the year they occur, not at some common point in time.

Uranium. The basic material for nuclear technology. It is a slightly radioactive naturally occurring heavy metal that is more dense than lead. Uranium is 40 times more common than silver.

Vitrification. The process of immobilizing waste that produces a glass-like solid that permanently captures the radioactive materials.

Vulnerabilities. Conditions or weaknesses that may lead to radiation exposure to the public, unnecessary or increased exposure to the workers, or release of radioactive materials to the environment.

Waste classification. Wastes are classified according to 10 CFR § 61.55 for the purpose of near surface disposal to three classes: A, B, and C. Class C waste represents the waste that must meet the most rigorous requirements on waste form to ensure stability and additional measures at the disposal facility to protect against inadvertent intrusion.

Waste management. The planning, coordination, and direction of those functions related to generation, handling, treatment, storage, transportation, and disposal of waste, as well as associated surveillance and maintenance activities.

Waste minimization. An action that economically avoids or reduces the generation of waste by source reduction or reduces the toxicity of hazardous waste, improving energy usage, or by recycling. This action will be consistent with the general goal of minimizing present and future threats to human health, safety, and the environment.

Wet storage. Storage of spent nuclear fuel in a pool of water, generally for the purposes of both cooling and worker shielding.