DOE/EA-1977

DRAFT

ENVIRONMENTAL ASSESSMENT

FOR THE

ACCEPTANCE AND DISPOSITION OF SPENT NUCLEAR FUEL CONTAINING U.S.-ORIGIN HIGHLY ENRICHED URANIUM FROM THE FEDERAL REPUBLIC OF GERMANY

January 2016

U.S. DEPARTMENT OF ENERGY

SAVANNAH RIVER OPERATIONS OFFICE

AIKEN, SOUTH CAROLINA

SUMMARY

S.1. Introduction

In accordance with the Council on Environmental Quality's (CEQ) National Environmental Policy Act (NEPA) regulations at 40 Code of Federal Regulations (CFR) Parts 1500 through 1508, Executive Order 12114, *Environmental Effects Abroad of Major Federal Actions*, and U.S. Department of Energy (DOE) NEPA implementing procedures at 10 CFR Part 1021, DOE has prepared this Environmental Assessment (EA) to evaluate the receipt, storage, processing and disposition of certain spent nuclear fuel from a research and development program of the Federal Republic of Germany (Germany).[1] DOE is considering the feasibility of accepting this spent nuclear fuel containing U.S.-origin highly enriched uranium[2] (HEU) at DOE's Savannah River Site (SRS) for processing and disposition. The United States provided the HEU to Germany between 1965 and 1988. DOE and Germany have signed a Statement of Intent (included as Appendix A to this EA) to cooperate in conducting preparatory work necessary to support DOE's consideration of the proposed use of SRS facilities for these activities. If DOE and Germany decide to proceed with the proposed action, the German government would be responsible for transporting the spent nuclear fuel from storage in Germany to the United States, at which point the United States would take responsibility for the spent fuel. The Statement of Intent specifies that Forschungszentrum Julich, an interdisciplinary research center funded primarily by the German government, is bearing the cost of the preparatory phase – feasibility studies and NEPA analysis – and if there is a decision to proceed with the project, would also bear the costs associated with acceptance, processing, and disposition of the spent nuclear fuel.

S.2 Background

The spent nuclear fuel that is the subject of this proposal was irradiated in two German reactors that operated as part of Germany's research and development program for pebble bed, high-temperature, gas-cooled reactor technology, the Arbeitsgemeinschaft Versuchsreaktor (AVR), which operated from 1967 to 1988; and the Thorium High Temperature Reactor-300 (THTR), which operated from 1983 to 1989. The AVR spent nuclear fuel has been stored in Jülich, Germany, and the THTR spent nuclear fuel has been stored in Ahaus, Germany, since the reactors were shut down and defueled.

This spent fuel is in the form of small graphite (carbon) spheres, referred to as "pebbles." There are approximately one million pebbles currently in storage in 455 CASTOR[3] casks. The pebbles contain varying quantities of uranium and thorium, with uranium enrichments up to 81 percent. Prior to irradiation, the fuel contained approximately 900 kilograms (1,980 pounds) of HEU

[1] This environmental assessment was announced as the *Environmental Assessment for the Acceptance and Disposition of Used Nuclear Fuel Containing U.S.-Origin Highly Enriched Uranium from the Federal Republic of Germany* in DOE's Notice of Intent (NOI) on June 4, 2014 (79 FR 32256).

[2] Highly enriched uranium has a concentration of 20 percent or greater of the isotope uranium-235. Natural uranium contains approximately 0.7 percent uranium-235.

[3] CASTOR is an abbreviation for "cask for storage and transport of radioactive material."

provided by the United States (Schütte 2012). As a result of irradiation and decay, the spent nuclear fuel also contains actinides, fission products, and other radioactive isotopes.

German Request to Return U.S.-Origin HEU. The United States has a policy objective to reduce, and eventually to eliminate, HEU from civil commerce. In February 2012, the German government approached DOE about the possibility of the United States accepting the spent nuclear fuel for storage and disposition (Schütte 2012). As a result of discussions, Germany funded Savannah River National Laboratory to conduct research that would lead to a method to separate the fuel kernels from the graphite matrix, the first step in processing this fuel. DOE agreed to consider Germany's request for the following reasons: the spent fuel contains U.S.-origin HEU; success of the above-mentioned research on a laboratory scale; SRS expertise in nuclear engineering and the management of nuclear materials; and availability of hardened SRS facilities that could be used as is or modified to process and disposition this type of spent nuclear fuel.

A Statement of Intent between DOE for the United States, the Ministry of Education and Research for the Federal Republic of Germany, and the Ministry for Innovation, Science, and Research for the State of North Rhine-Westphalia (on behalf of the North Rhine-Westphalian State Government), was signed in late March and early April 2014. The Statement of Intent enabled DOE and the German signatories to continue evaluating the feasibility of this proposed project, and to conduct additional studies and reviews required to determine whether to proceed with acceptance of the spent nuclear fuel for processing and disposition, including the preparation of this EA.

Future development activities to advance the technology will involve several major maturation activities. These include remote opening and handling of the CASTOR casks, design of a fully-integrated prototypical digestion system, operation of prototypical equipment in a remote-handle configuration, and obtaining critical process data using irradiated fuel kernels and individual pebbles. The maturation approach will also address essential safety, security, and facility interface issues which include facility permitting, waste disposal, and final fuel disposition. All of these research activities are being conducted under Categorical Exclusion B3.6 (Small-Scale Research and Development, Laboratory Operations, and Pilot Projects documented in a series of Categorical Exclusion Determinations prepared by the SRS NEPA Compliance Officer [DOE 2013f, 2013g, 2014f, 2015b, 2015c]). Should future research and development requirements be different from those evaluated and approved in these evaluations, additional NEPA reviews will be conducted prior to initiating those activities.

Savannah River Site Capabilities. The facilities and capabilities proposed for processing this spent nuclear fuel are unique to DOE and SRS. H-Canyon, which began operating in 1955, is the only hardened nuclear chemical separations plant still in operation in the United States. H-Canyon continues to be used to separate and recover uranium from spent nuclear fuel and other highly radioactive materials for reuse and to prepare the residuals for disposal through the SRS Liquid Nuclear Waste Facilities.

L-Area was initially constructed as a nuclear reactor for use as a nuclear material production facility in the 1950s. The reactor was permanently shut down in the 1980s, but the ancillary facilities have continued to support SRS missions. In the early 2000s, research and development was conducted at SRS for the melt and dilute technology, a method for stabilizing spent nuclear fuel that is now proposed under the L-Area Alternative (see Section S.6). During that time frame, conceptual design for implementation of the melt and dilute technology in L-Area facilities was initiated but later halted.

The SRS Liquid Nuclear Waste Facilities are an extensive, integrated processing and disposition system comprising several facilities and technologies that do not exist elsewhere in the United States. The Liquid Nuclear Waste Facilities include storage, processing, and disposal facilities: tank farms, the Defense Waste Processing Facility, saltstone facilities, and existing and planned glass waste storage facilities.

S.3 Purpose and Need

DOE's purpose and need for the receipt, storage, processing, and disposition of the spent nuclear fuel from Germany is to support the U.S. policy objective to reduce, and eventually to eliminate, HEU from civil commerce (White House 1993). This action would further the U.S. HEU minimization objective by returning U.S.-origin HEU[4] from Germany to the United States for safe storage and disposition in a form no longer usable for an improvised nuclear device, a radiological dispersal device, or other radiological exposure device.

S.4 Proposed Action

If the current feasibility studies show adequate promise, and DOE and Germany decide to proceed with the project, the German government would work with DOE to transport spent nuclear fuel in chartered ships across the Atlantic Ocean to Joint Base Charleston-Weapons Station, near Charleston, South Carolina. From Joint Base Charleston-Weapons Station, the casks would be transported to SRS on dedicated trains in accordance with applicable U.S. regulatory requirements. **Figure S-1** shows the locations of facilities for the proposed activities.

[4] Prior to irradiation, the fuel contained approximately 900 kilograms (1,980 pounds) of HEU (Schütte, 2012).

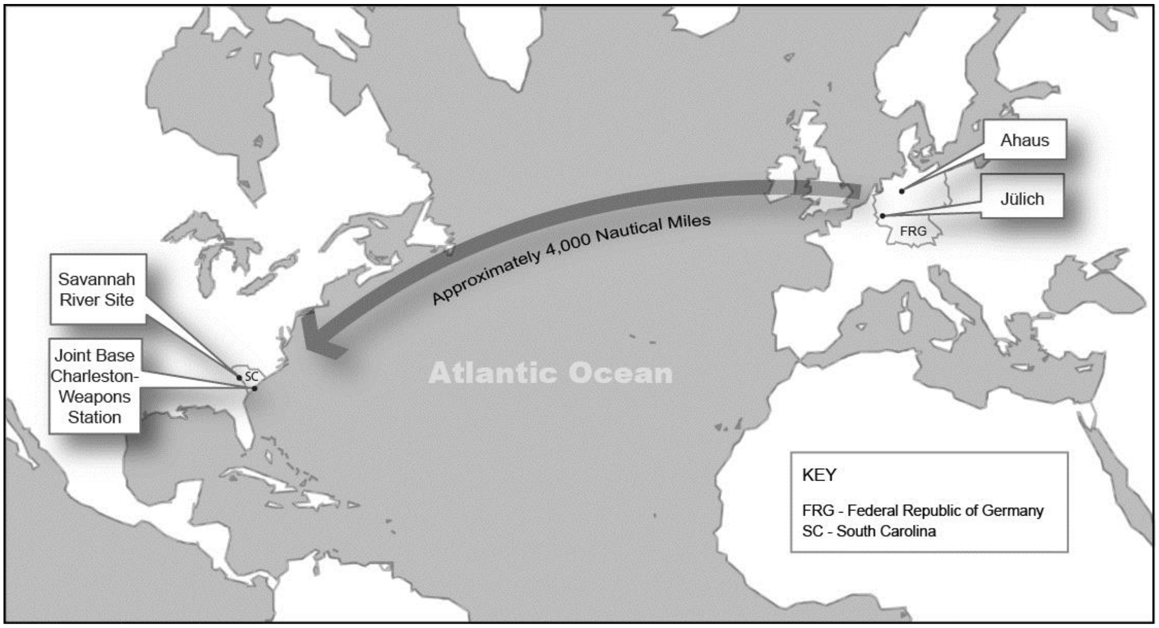

Figure S-1: Proposed Project Locations

The spent fuel would be stored at SRS in CASTOR casks, the Type B transportation casks[5] in which it would be shipped, until installation of the new equipment needed for initial processing of the spent nuclear fuel is completed. SRS infrastructure and facilities in E-Area, H-Area (including H-Canyon), and L-Area, as well as the Liquid Nuclear Waste Facilities would be used to process the spent nuclear fuel from Germany. Alternatives for implementing the Proposed Action, including the facilities required, are described in Section S.6.

As specified in the Statement of Intent, any decision by the Participants (signatories to the Statement of Intent) to proceed with the transportation of the spent fuel for acceptance, processing, and disposition depends on compliance with all applicable requirements of United States law and DOE requirements, including NEPA, and resolution by the Participants of any technical, financial, and legal issues that may be identified during consideration of the feasibility of the project and development of an appropriate legal framework.

S.5 Public Involvement

DOE announced its intent to prepare this Draft EA with publication of a Notice of Intent in the *Federal Register* on June 4, 2014 (79 FR 32256). DOE invited Federal agencies, state and local governments, Native American tribes, industry, other organizations, and members of the public to submit comments on the proposed scope of the EA. The public scoping period opened with

[5] Type B packages are required for the transport of highly radioactive material. Type B packages must withstand, without loss of contents, normal transport conditions such as heat, cold, vibration, changes in pressure, being dropped, compressed, sprayed with water, or struck by objects, as well as more serious accident conditions. These requirements are demonstrated during the licensing process for each Type B package through rigorous testing in accordance with 10 CFR 71, Packaging and Transportation of Radioactive Material.

publication of the Notice of Intent, and closed on July 21, 2014. DOE has continued to accept and consider comments throughout preparation of this Draft EA. A public Scoping Meeting was held on June 24, 2014, at the North Augusta Community Center, North Augusta, South Carolina.

Approximately 227 public comment documents, including those in two letter campaigns, have been received since the public scoping period opened. Comments both in support of and opposed to the proposed project were received, as well as requests for details about how the project would be implemented, for specific analyses, and for assurances that Germany would pay the full cost of the project. DOE considered all comments received in developing the alternatives to be evaluated and in preparing this EA. A summary of the comments received and DOE's response to those comments is in Section 1.5 of this EA.

S.6 Description of Alternatives for Acceptance and Disposition of Spent Nuclear Fuel from Germany

DOE is evaluating two alternatives for acceptance and disposition of graphite-based spent nuclear fuel currently stored in Germany, and, as required by DOE's NEPA implementing procedures (10 CFR 1021.321(c)), a No Action Alternative. Under the No Action Alternative, the spent fuel would not be transported to the United States for management and disposition.

Under the action alternatives, the spent nuclear fuel would be transported from Germany and processed at SRS for final disposition as a proliferation-resistant waste form. The two action alternatives differ in processing technology and location at SRS where the processing would occur. The H-Area Alternative (so named because most activities would involve H-Area facilities) has three processing options (Vitrification Option, Low-Enriched Uranium[6] (LEU) Waste Option, and LEU/Thorium Waste Option) that use H-Canyon to differing extents; the L-Area Alternative (so named because the alternative would involve mostly L-Area facilities) would use a melt and dilute process in L-Area. The action alternatives and the associated processing options are shown in **Figure S–2**.

[6] Low-enriched uranium has a concentration of the isotope uranium-235 above that of natural uranium (0.7 percent), but less than 20 percent.

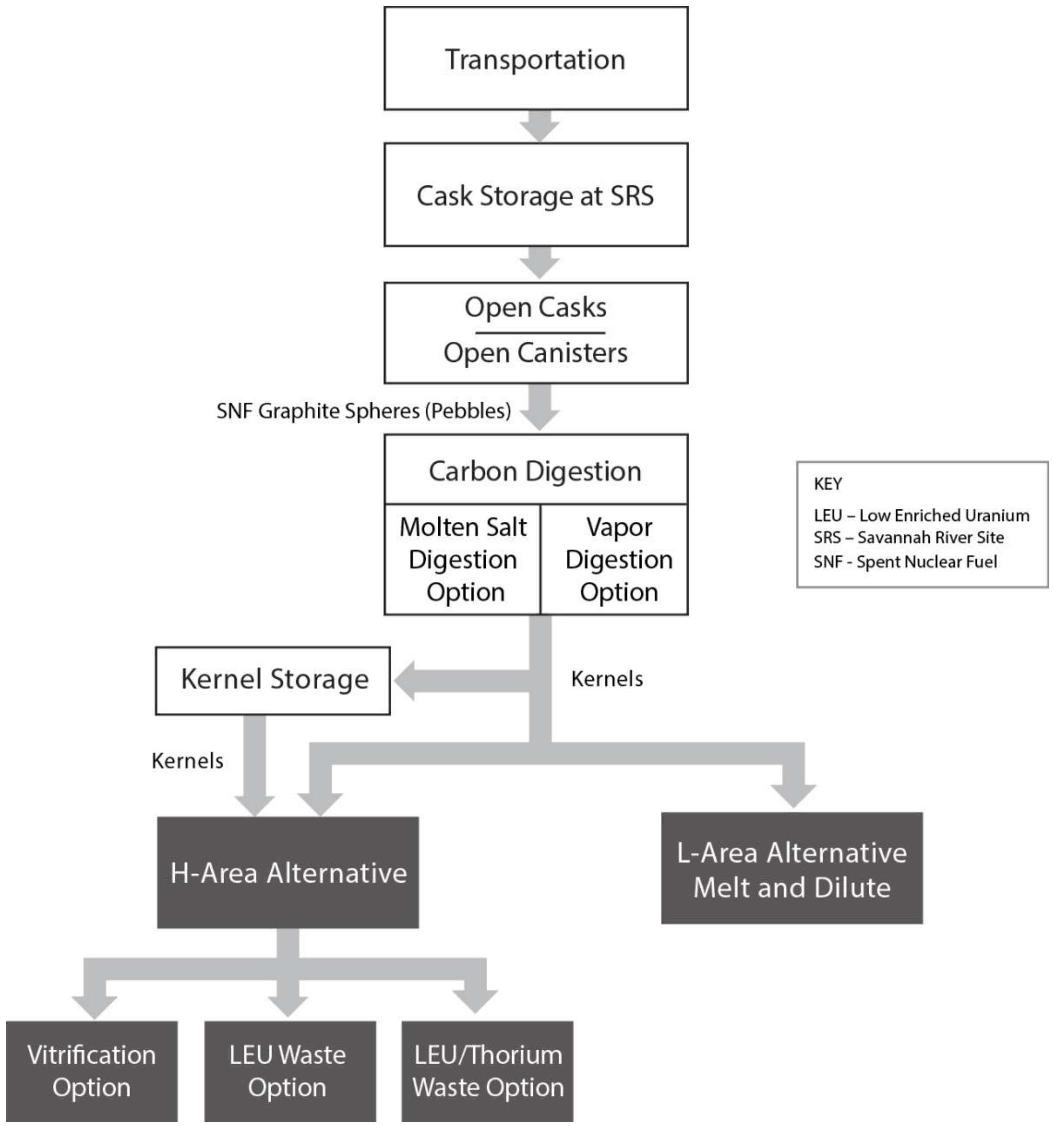

Figure S-2: H-Area and L-Area Alternatives

The German government would place the CASTOR casks into shipping containers and transport them from the Jülich and Ahaus sites to a seaport in northern Germany where they would be secured aboard chartered ships certified to carry nuclear material. Consistent with Executive Order 12114, *Environmental Effects Abroad of Major Federal Actions*, the environmental impacts analysis in this EA starts at the point of the transport ships entering the global commons.[7]

The shipping campaign from Germany would involve about 30 shipments over approximately a 3.5-year period to transport the 455 CASTOR casks of spent nuclear fuel from Germany; a typical shipment would include 16 casks. At Joint Base Charleston-Weapons Station, railcars would be

[7] Global commons refers to areas that are outside the jurisdiction of any nation (e.g., the oceans or Antarctica).

staged in advance of the arrival of the ship at the dock. Transport to SRS would be by a commercial carrier using a dedicated train. The National Nuclear Security Administration infrastructure and protocols for receipt of foreign research reactor spent nuclear fuel would be followed for these shipments, including Federal and State coordination protocols, and those for transport, security, and radiation control.

The CASTOR casks containing the spent nuclear fuel from Germany would be offloaded from the rail cars at SRS and stored on existing and/or new concrete or gravel storage pads in H-Area, L-Area, or a combination of the areas. Upon receipt, the shipment would be subject to visual inspection, radiological survey, and data verification to ensure that the casks meet all acceptance requirements.

The preliminary processing steps, from removing the pebbles from the casks through carbon digestion (white boxes in Figure S-2), but not the facilities in which the activities occur, are the same for both the H-Area Alternative and the L-Area Alternative. After carbon digestion, the processing steps for the two alternatives diverge (shaded boxes in Figure S-2). The H-Area and L-Area candidate facilities considered for processing have robust structural features, established perimeter security zones, and sufficient area for cask storage and staging or construction of new facilities, if needed.

The HEU kernels are embedded in a graphite (carbon) matrix that must be removed for the HEU kernels to be processed. Two methods for removing the graphite surrounding the fuel kernels (referred to as carbon digestion) are under consideration: a molten salt digestion process and a vapor digestion process. Both of the carbon digestion methods are evaluated in this EA for implementation in either H- or L-Areas.

Four kernel processing options are being considered. Three options under the H-Area Alternative (these options would be implemented in H-Area) and one option under the L-Area Alternative that would be installed in a modified wing of the L-Area Material Storage Facility (Building 105-L). The four options for processing the kernels after carbon digestion are:

H-Area Alternative Options:

- **Vitrification Option –** Dissolution of the kernels in H-Canyon with direct transfer of the dissolver solution to the existing Liquid Nuclear Waste Facilities.
- **LEU Waste Option** – Dissolution of the kernels in H-Canyon followed by solvent extraction in H-Canyon for separation of the uranium. The uranium solution would be down blended and grouted (i.e., solidified by mixing with cement) to meet acceptance criteria for disposal as low-level radioactive waste (LLW). Thorium, other actinides, and fission products would be processed through the Liquid Nuclear Waste Facilities.
- **LEU/Thorium Waste Option** – Dissolution of the kernels in H-Canyon followed by solvent extraction in H-Canyon for separation of the uranium and thorium. The uranium/thorium solution would be down blended and grouted to meet acceptance criteria for disposal as LLW. Other actinides and fission products would be processed through the Liquid Nuclear Waste Facilities.

L-Area Alternative Option:

- **Melt and Dilute Option:** Down blending and conversion of the kernels to a uranium-aluminum alloy in a melt and dilute process in L-Area. The resulting ingots would be stored in concrete overpacks on a pad in L-Area. Unlike the H-Area processing methods, the kernels would not be dissolved prior to final processing.

Some modifications to the interiors of existing facilities (specifically, H-Canyon or the L-Area Material Storage Facility) would be required to implement any of these alternatives or options. In addition, construction of storage pads for cask storage and minor onsite road construction could be required, depending on the alternative. For the H-Area Alternative, LEU Waste and LEU/Thorium Waste Options, a separate uranium solidification building in H-Area would be constructed. For the L-Area Alternative, a sand filter, fan room, stack, and truck bay would be built in L-Area. Processing, from kernel dissolution through production of the final waste form would take slightly less than 5 years for the H-Area Alternative Vitrification Option, and approximately 5 years for the LEU and LEU/Thorium Options. Processing for the L-Area Melt and Dilute Alternative would take approximately 7 years**.**

S.7 Summary of Environmental Consequences

No Action Alternative

The SNF containing U.S.-origin HEU from the AVR and THTR reactors would remain in storage in Germany. It would not be transported to the United States for management and disposition. Because DOE would not undertake any actions involving the global commons, Joint Base Charleston–Weapons Station, or SRS under the No Action Alternative, there would be no additional impacts on these areas.

Action Alternatives

Global Commons and Joint Base Charleston –Weapons Station. Because of the small number of shipments (about 30 over an approximately 3.5-year period, as compared to the several thousand vessels that annually traverse the global commons and the 35 to 45 vessels[8] that are received annually at Joint Base Charleston – Weapons Station) and environmental laws, regulations, and best practices, nonradiological impacts on the global commons and Joint Base Charleston – Weapons Station from shipment of spent nuclear fuel from Germany are expected to be minimal. The public would not receive a radiation dose from incident-free ocean transport or unloading at Joint Base Charleston – Weapons Station. The total radiation dose among all ship crew members from ocean transport of the fuel would be 2.9 person-rem. No latent cancer fatalities (LCFs) would be expected (calculated value of 2×10^{-3}) as a result of this collective dose. The total dose among all workers from unloading at Joint Base Charleston – Weapons Station is projected to be approximately 0.24 person-rem, with no LCFs expected from this dose (calculated value of 1×10^{-4} LCF).

Radiological Impacts

In this EA, radiological consequences of operations and accidents are reported as doses and latent cancer fatalities (LCFs). An LCF is a death from cancer resulting from, and occurring some time after, exposure to ionizing radiation. A factor of 0.0006 LCFs per rem or person-rem is used to calculate the risk associated with radiation doses (DOE 2003); for acute individual doses above 20 rem, the risk factor is doubled (NCRP 1993).

For a group (for example, the offsite population), doses are reported in person-rem and LCFs are reported as a whole number, representing the number of people in the group statistically expected to develop an LCF as a result of the exposure. When the value calculated by multiplying the dose by the LCF risk factor of 0.0006 is less than 1, the reported value is rounded to 0 or 1 and the calculated value is shown in parentheses. For an individual, doses are reported in rem or millirem, along with the risk or likelihood of the dose resulting in an LCF. Because it is assumed that there is some level of risk associated with radiation exposure, regardless of the magnitude, the individual risk is not reported as 0.

The probability of an accident that could result in a CASTOR cask being submerged in coastal waters was estimated to be 2.9×10^{-11} (1 in 35 billion) for a damaged cask, and 1.5×10^{-8} (1 in 67 million) for an undamaged cask. The probability of an accident that could result in a CASTOR cask being submerged in deep ocean waters was estimated to be 1.1×10^{-6} (1 in

[8] Joint Base Charleston – Weapons Station is able to handle this potential increase in vessels and would provide the staff necessary for safe unloading operations.

910,000) (the cask was assumed to be damaged). The probabilities of accidents at Joint Base Charleston–Weapons Station that could release radioactivity are expected to range from 6.5×10^{-6} (1 in 150,000) to 6.0×10^{-10} (1 in 1.7 billion) with no population LCFs (calculated values: 3×10^{-6} to 3×10^{-2}) expected. The total risk of an LCF in the population due to a spent nuclear fuel from Germany accident at Joint Base Charleston–Weapons Station is estimated at 9.8×10^{-8}.

Savannah River Site. Table S–1 summarizes the potential impacts at SRS for those resource areas having the greatest potential for environmental impacts (Air Quality, Human Health, Socioeconomics, Waste Management, Transportation, and Environmental Justice) for the action alternatives evaluated in this German Fuel EA. Activities related to the evaluated alternatives would largely occur in existing industrial areas far from offsite areas. In addition, little land would be disturbed, contaminated water would not be discharged, and resource use would be low. Therefore, minimal or no impacts are expected to the other resources areas regardless of the alternative.

Cumulative Impacts. Incident-free ocean transport of spent nuclear fuel from Germany would not result in radiation exposures to members of the general public. Therefore, there would be no cumulative radiation impact to members of the general public. Cumulative radiation doses and risks to ship crews and dock handlers from transport of radioactive materials from foreign countries to U.S. seaports would result in a dose of 89-person-rem and no LCFs (calculated value of 5×10^{-2}). Shipments of the spent nuclear fuel from Germany would represent approximately 4 percent of the cumulative dose and risk resulting from all shipments of radioactive materials from foreign countries to U.S. seaports.

Because construction activities at SRS would be minor and small areas of land would be disturbed, air quality impacts would be minor and are not likely to contribute substantially to cumulative impacts. Because the operation of facilities for processing spent nuclear fuel from Germany would produce relatively small quantities of criteria air pollutants and hazardous air pollutants, these emissions are not likely to contribute substantially to cumulative impacts.

The annual cumulative dose from SRS and offsite sources to the regional population is estimated to be 26 to 32 person-rem. This population dose is not expected to result in any LCFs (calculated value of 0.02). The annual contribution to the cumulative population dose from activities evaluated in this EA would be 7.3 to 7.8 person-rem for the H-Area Alternative and 2.3 person-rem for the L-Area Alternative, with no associated LCFs for either alternative (calculated values of 4×10^{-3} to 5×10^{-3} and 1×10^{-3} respectively). For perspective, the annual doses to the same population from naturally occurring radioactive sources (311 millirem per person) would be about 270,000 person-rem, from which approximately 160 LCFs would be inferred. The cumulative annual SRS worker dose from current and reasonably foreseeable activities is estimated to be 840 to 870 person-rem, which is not expected to cause an LCF (calculated value of 0.5) among the involved worker population. Activities evaluated in this EA could result in annual worker doses of 28 to 41 person-rem for the H-Area Alternative and 8 person-rem for the L-Area Alternative with no associated LCFs for either alternative (calculated values of 0.02 and 0.005, respectively).

The construction, modification, and operation of SRS facilities that DOE would use to disposition the spent nuclear fuel from Germany are not expected to impact resources associated with current or future site activities, remediation efforts or site closure. Because Germany would pay for disposition of the spent nuclear fuel from Germany, U.S. government funding for other SRS projects would not be affected. DOE would modify existing facilities to implement the alternatives.

However, these activities would not impact future decommissioning, decontamination and demolition efforts since they are a small subset of the activities at the facilities being impacted. The new uranium solidification facility would be designed to facilitate decommission, decontamination and demolition. The waste volumes that would be generated from decontamination and demolition would be a small fraction of those from decontamination and demolition of existing facilities and would likely be performed concurrently. The scheduled timeframe for closure of the HLW tanks is FY2039 (SRR 2014b), many years after completion of the German spent nuclear fuel project. Therefore, the impacts on site closure, if any, would be the additional time for disposing of the wastes associated with decommissioning, decontamination and demolition of the German fuel facilities. As described in Section 4.3.2.1, DOE anticipates that the impacts would be on the order of a few months to a year.

Table S-1: Summary Comparison of Environmental Consequences at SRS

Resource Area / Parameter		Action Alternative [a]			
		H-Area Alternative			L-Area Alternative
		Vitrification Option	LEU Waste Option	LEU/Thorium Waste Option	Melt and Dilute Option
Air Quality	***Construction***				
	Criteria Air Pollutant Emissions:	Emissions not expected to exceed existing permit levels	Same as Vitrification Option	Same as Vitrification Option	Same as H-Area Alternative, Vitrification Option
	Operations				
	Criteria Air Pollutant Emissions: HAPs: CEQ Draft GHG Guidance – 25,000 metric tons CO_{2e} reference point for quantitative analysis (CEQ 2014)	Increase in nitrogen dioxide emissions would require a permit review to determine whether revisions to the Title V Air Operating Permit would be required HAPs emitted in small quantities GHG emissions would be a marginal increase over the no action alternative	Same as Vitrification Option	Same as Vitrification Option	Increase in L-Area emissions may require a permit revision HAPs emitted in small quantities GHGs below reference point
Human Health – Normal Operations, Workers	***Construction***				
	Total Worker Dose (person-rem) Total Worker LCFs [b]	50 0 (0.03)	Same as Vitrification Option	Same as Vitrification Option	Work would not be performed in a radiation area; meaningful doses would not be expected
	Operations				
	Total Worker Dose (person-rem) Total Worker LCFs [b]	69 0 (0.04)	61 0 (0.04)	Same as LEU Waste Option	43 0 (0.03)
Human Health – Normal Operations, General Population	***Construction***				
	Radiological Exposure to the Public	None expected	Same as Vitrification Option	Same as Vitrification Option	Same as H-Area Alternative Vitrification Option
	Operations				
	Annual Population Dose (person-rem) Annual Population LCFs [b] Total Project Population LCFs [b] Annual MEI Dose (millirem) Annual MEI LCF Risk Total Project MEI LCF Risk	7.3 0 (0.004) 0 (0.01) 0.084 5×10^{-8} 1×10^{-7} Risk to the public would be small	7.8 0 (0.005) 0 (0.01) 0.12 6×10^{-8} 1×10^{-7} Risk to the public would be small	7.6 0 (0.005) 0 (0.01) 0. 012 6×10^{-8} 1×10^{-7} Risk to the public would be small	2.3 0 (0.001) 0 (0.009) 0.029 2×10^{-8} 1×10^{-7} Risk to the public would be small

Resource Area / Parameter		Action Alternative [a]			
		H-Area Alternative			L-Area Alternative
		Vitrification Option	LEU Waste Option	LEU/Thorium Waste Option	Melt and Dilute Option
Human Health – Facility Accidents	Operational Accident Frequency[c] Consequences Population LCFs MEI LCF Risk Beyond-Design-Basis Accident Frequency[c] Consequences Population LCFs MEI LCF Risk	<u>SNF Processing Accident</u> Extremely unlikely 47 8×10^{-4} <u>Earthquake with fire</u> Beyond extremely unlikely Not evaluated 0.1	Same as Vitrification Option	Same as Vitrification Option	<u>Melter fire</u> Extremely unlikely Not evaluated 8×10^{-4} <u>Earthquake induced spill</u> Beyond Extremely unlikely 13 3×10^{-4}
Socioeconomics	***Construction***				
	Peak Direct Employment Percent of SRS Employment	Up to 100 1.4 No noticeable impact.	Up to 201 2.8 No noticeable impact.	Same as LEU Waste Option.	Up to 155 2.1 No noticeable impact.
	Operations				
	Peak Direct Employment Percent of SRS Employment	125 to 150 1.7 to 2.1 No new jobs. Small beneficial impact by preserving existing jobs.	125 to 150 1.7 to 2.1 Most would be existing employees: as many as 20 new jobs for uranium solidification facility. Small beneficial impact by preserving existing jobs.	Same as LEU Waste Option.	135 1.9 No new jobs. Small beneficial impact by preserving existing jobs.
Waste Management (The values in parenthesis represent the percent of SRS waste management facility capacity)	***Construction***				
	Solid LLW (cubic meters) Solid Hazardous (cubic meters) Liquid Hazardous (liters) Solid Nonhazardous (cubic meters: Liquid Nonhazardous (liters)	320 (0.1) 0.15 (0.02) 190 (0.02) 110 (0.0009) 9,500 (2×10^{-4}) Waste management capacities are sufficient for these waste streams.	320 (0.1) 1.7 (0.3) 570 (0.1) 340 (0.004) 32,000 (0.001) Waste management capacities are sufficient for these waste streams.	Same as LEU Waste Option	390 (0.1) NG NG NG NG Waste management capacities are sufficient for these waste streams.
	Operations				
	Solid LLW (cubic meters) Liquid LLW (liters) Solid Hazardous (cubic meters) Solid Nonhazardous (cubic meters) Liquid Nonhazardous (liters) HLW Canisters (number) Saltstone Grout (liters):	2,000 (0.7) NG NG NG NG 101 (2) 5,500,000 (16 - 24)[e] Waste management capacities are sufficient for these waste streams.	2,300 (0.8) 280,000 (0.03) 0.15 (0.03) 75 (0.001) 2,800,000 (0.1) 32 (0.7) 6,200,000 (18 to 27) [e] Waste management capacities are sufficient for these waste streams.	2,600 to 2,900 (0.9 to 1.0) 280,000 (0.03) 0.15 (0.03) 75 (0.001) 2,800,000 (0.1) 15 (0.3) 6,200,000 (18 to 27) [e] Waste management capacities are sufficient for these waste streams.	2,000 (0.7) NG NG NG NG 82 (NA [d]) 3,700,000 (5 to 8)[e] Waste management capacities are sufficient for these waste streams.

Resource Area / Parameter		Action Alternative [a]			
		H-Area Alternative			L-Area Alternative
		Vitrification Option	LEU Waste Option	LEU/Thorium Waste Option	Melt and Dilute Option
Transportation (total health effects)	Shipments	30	330	540	30
	Incident-free				
	- Crew LCF risk	7×10^{-5}	4×10^{-3}	7×10^{-3}	7×10^{-5}
	- Population LCF risk	3×10^{-4}	2×10^{-3}	3×10^{-3}	3×10^{-4}
	Accidents				
	Population LCF Risk	5×10^{-13}	5×10^{-6}	5×10^{-6}	5×10^{-12}
	Traffic fatalities	9×10^{-4}	5×10^{-2}	9×10^{-2}	9×10^{-4}
Environmental Justice	***Construction***				
	Impacts on minority or low-income populations	No disproportionately high and adverse impacts on minority or low-income populations are expected.			
	Operations				
	Impacts on minority or low-income populations	No disproportionately high and adverse impacts on minority or low-income populations are expected.			

CEQ = Council on Environmental Quality; DWPF = Defense Waste Processing Facility; GHG = greenhouse gas; HAP = hazardous air pollutant; HLW = high-level radioactive waste; LCF = latent cancer fatality; LEU = low-enriched uranium; LLW = low-level radioactive waste; MEI = maximally exposed (offsite) individual; NA = not applicable; NG = not generated in meaningful quantities; SNF = spent nuclear fuel; SRS = Savannah River Site.

[a] Under the No Action Alternative, the spent nuclear fuel from Germany would not be transported to the United States for management and disposition. The SNF would remain in storage in Germany. Because DOE would not undertake any actions under the No Action Alternative, there would be no incremental impacts at SRS.

[b] The number of excess LCFs in the population would occur as a whole number. If the number is zero, the value calculated by multiplying the dose by a risk factor of 0.0006 LCF per person-rem (DOE 2003) is presented in parenthesis.

[c] Frequencies are on an annual basis and defined as: extremely unlikely = 10^{-6} to 10^{-4}, beyond extremely unlikely = less than 10^{-6}.

[d] Capacity for HLW canisters under this German Fuel EA is determined by comparison with storage capacity at the S-Area Glass Waste Storage Buildings. However, multi-canister overpacks from melt and dilute operations at L-Area would be stored on an L-Area pad rather than at S-Area.

[e] The quantity of saltstone grout is the total for the project duration (approximately 3.5 years for the H-Area Alternative and approximately7 years for the L-Area Alternative); however, the percent of capacity (value in parenthesis) is based on the annual saltstone processing rate.

Notes: To convert cubic meters (solid) to cubic yards, multiply by 1.3079; cubic meters (liquid) to cubic feet, multiply by 35.314; liters to gallons, multiply by 0.26418; acres to hectares, multiply by 0.40469.

Source: DOE 2014a

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ABBREVIATIONS and ACRONYMS

AVR	Arbeitsgemeinschaft Versuchsreaktor
ACS	American Community Survey
BMP	best management practice
CEQ	Council on Environmental Quality
CERCLA	Comprehensive Environmental Response, Compensation and Liability Act
CFR	Code of Federal Regulations
CSWTF	Central Sanitary Wastewater Treatment Facility
DOE	U.S. Department of Energy
DWPF	Defense Waste Processing Facility
EA	environmental assessment
EPA	U.S. Environmental Protection Agency
FY	fiscal year
HLW	high-level radioactive waste
HEU	highly enriched uranium
IMO	International Maritime Organization
LCF	latent cancer fatality
LEU	low-enriched uranium
LLW	low-level radioactive waste
MAR	material at risk
MCO	multi-canister overpack
NEPA	National Environmental Policy Act
NESHAPS	National Emission Standards for Hazardous Air Pollutants
NMFS	National Marine Fisheries Service
NPDES	National Pollutant Discharge Elimination System
NRCS	Natural Resource Conservation Service
OSHA	Occupational Safety and Health Administration
PGA	peak ground acceleration
RCRA	Resource Conservation and Recovery Act
rem	Roentgen equivalent man
RIMS II	Regional Input and Output Modeling System
ROI	region of influence
SCDHEC	South Carolina Department of Health and Environmental Control
SNF	spent nuclear fuel (also called used nuclear fuel)
SRS	Savannah River Site
SWPPP	Stormwater Pollution Prevention Plan
THTR	Thorium High Temperature Reactor - 300
U.S.	United States
USDA	U.S. Department of Agriculture

CONVERSIONS

METRIC TO ENGLISH			ENGLISH TO METRIC		
Multiply	**by**	**To get**	**Multiply**	**by**	**To get**
Area					
Square meters	10.764	Square feet	Square feet	0.092903	Square meters
Square kilometers	247.1	Acres	Acres	0.0040469	Square kilometers
Square kilometers	0.3861	Square miles	Square miles	2.59	Square kilometers
Hectares	2.471	Acres	Acres	0.40469	Hectares
Concentration					
Kilograms/square meter	0.16667	Tons/acre	Tons/acre	0.5999	Kilograms/square meter
Milligrams/liter	1 [a]	Parts/million	Parts/million	1 [a]	Milligrams/liter
Micrograms/liter	1 [a]	Parts/billion	Parts/billion	1 [a]	Micrograms/liter
Micrograms/cubic meter	1 [a]	Parts/trillion	Parts/trillion	1 [a]	Micrograms/cubic meter
Density					
Grams/cubic centimeter	62.428	Pounds/cubic feet	Pounds/cubic feet	0.016018	Grams/cubic centimeter
Grams/cubic meter	0.0000624	Pounds/cubic feet	Pounds/cubic feet	16,018.5	Grams/cubic meter
Length					
Centimeters	0.3937	Inches	Inches	2.54	Centimeters
Meters	3.2808	Feet	Feet	0.3048	Meters
Kilometers	0.62137	Miles	Miles	1.6093	Kilometers
Radiation					
Sieverts	100	Rem	Rem	0.01	Sieverts
Temperature					
Absolute					
Degrees C + 17.78	1.8	Degrees F	Degrees F - 32	0.55556	Degrees C
Relative					
Degrees C	1.8	Degrees F	Degrees F	0.55556	Degrees C
Velocity/Rate					
Cubic meters/second	2118.9	Cubic feet/minute	Cubic feet/minute	0.00047195	Cubic meters/second
Grams/second	7.9366	Pounds/hour	Pounds/hour	0.126	Grams/second
Meters/second	2.237	Miles/hour	Miles/hour	0.44704	Meters/second
Volume					
Liters	0.26418	Gallons	Gallons	3.7854	Liters
Liters	0.035316	Cubic feet	Cubic feet	28.316	Liters
Liters	0.001308	Cubic yards	Cubic yards	764.54	Liters
Cubic meters	264.17	Gallons	Gallons	0.0037854	Cubic meters
Cubic meters	35.314	Cubic feet	Cubic feet	0.028317	Cubic meters
Cubic meters	1.3079	Cubic yards	Cubic yards	0.76456	Cubic meters
Cubic meters	0.0008107	Acre-feet	Acre-feet	1233.49	Cubic meters
Weight/Mass					
Grams	0.035274	Ounces	Ounces	28.35	Grams
Kilograms	2.2046	Pounds	Pounds	0.45359	Kilograms
Kilograms	0.0011023	Tons (short)	Tons (short)	907.18	Kilograms
Metric tons	1.1023	Tons (short)	Tons (short)	0.90718	Metric tons
ENGLISH TO ENGLISH					
Acre-feet	325,850.7	Gallons	Gallons	0.000003046	Acre-feet
Acres	43,560	Square feet	Square feet	0.000022957	Acres
Square miles	640	Acres	Acres	0.0015625	Square miles

a. This conversion is only valid for concentrations of contaminants (or other materials) in water.

METRIC PREFIXES

Prefix	Symbol	Multiplication factor
exa-	E	1,000,000,000,000,000,000 = 10^{18}
peta-	P	1,000,000,000,000,000 = 10^{15}
tera-	T	1,000,000,000,000 = 10^{12}
giga-	G	1,000,000,000 = 10^{9}
mega-	M	1,000,000 = 10^{6}
kilo-	k	1,000 = 10^{3}
deca-	D	10 = 10^{1}
deci-	d	0.1 = 10^{-1}
centi-	c	0.01 = 10^{-2}
milli-	m	0.001 = 10^{-3}
micro-	µ	0.000 001 = 10^{-6}
nano-	n	0.000 000 001 = 10^{-9}
pico-	p	0.000 000 000 001 = 10^{-12}

1 INTRODUCTION AND PURPOSE AND NEED

1.1 INTRODUCTION

In accordance with the Council on Environmental Quality's National Environmental Policy Act (NEPA) regulations at 40 Code of Federal Regulations (CFR) Parts 1500 through 1508, Executive Order 12114, *Environmental Effects Abroad of Major Federal Actions*, and U.S. Department of Energy (DOE) NEPA implementing procedures at 10 CFR Part 1021, DOE has prepared this Environmental Assessment (EA) to evaluate the receipt, storage, processing, and disposition of certain spent nuclear fuel from a research and development program of the Federal Republic of Germany (Germany).[9] DOE is considering the feasibility of accepting this spent nuclear fuel at DOE's Savannah River Site (SRS) for processing and disposition. This spent fuel contains U.S.-origin highly enriched uranium[10] (HEU) provided to Germany between 1965 and 1988. DOE and Germany have signed a Statement of Intent (Appendix A) to cooperate in conducting preparatory work necessary to support DOE's consideration of the proposed use of facilities at SRS for these activities. If DOE and Germany decide to proceed with transportation of the spent nuclear fuel for storage, processing, and disposition, the German government would be responsible for transporting the spent fuel from storage in Germany to the United States, at which point the United States would take responsibility for the spent fuel. The Statement of Intent specifies that Forschungszentrum Julich, an interdisciplinary research center funded primarily by the German government, is bearing the cost of the preparatory phase – feasibility studies and NEPA analysis – and if there is a decision to proceed with the project, would also bear the costs associated with acceptance, processing, and disposition of the spent nuclear fuel.

1.2 BACKGROUND

The spent nuclear fuel that is the subject of this proposal was irradiated in two German reactors that operated as part of Germany's research and development program for pebble bed, high-temperature, gas-cooled reactor technology, the Arbeitsgemeinschaft Versuchsreaktor (AVR), which operated from 1967 to 1988; and the Thorium High Temperature Reactor-300 (THTR), which operated from 1983 to 1989. The AVR spent nuclear fuel has been stored in Jülich, Germany, and the THTR spent nuclear fuel has been stored in Ahaus, Germany, since the reactors were shut down and defueled.

This spent nuclear fuel is in the form of small graphite (carbon) spheres, referred to as pebbles. There are approximately one million pebbles currently in storage in CASTOR[11] casks. The pebbles contain varying quantities of uranium and thorium, with uranium enrichments up to 81 percent. Prior to irradiation, the fuel contained approximately 900 kilograms (1,980 pounds) of HEU provided by the United States (Schütte 2012). As a result of irradiation and decay, the

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[9] This environmental assessment was announced as the *Environmental Assessment for the Acceptance and Disposition of Used Nuclear Fuel Containing U.S.-Origin Highly Enriched Uranium from the Federal Republic of Germany* in DOE's Notice of Intent (NOI) on June 4, 2014 (79 FR 32256).

[10] Highly enriched uranium has a concentration of 20 percent or greater of the isotope uranium-235. Natural uranium contains approximately 0.7 percent uranium-235.

[11] CASTOR is an abbreviation for "cask for storage and transport of radioactive material."

spent nuclear fuel also contains actinides, fission products, and other radioactive isotopes.

German Request to Return U.S.-Origin HEU. The United States has a policy objective to reduce, and eventually to eliminate, HEU from civil commerce. In February 2012, the German government approached DOE about the possibility of the United States accepting the spent nuclear fuel for storage and disposition (Schütte 2012). DOE's Office of Environmental Management, which has the lead responsibility for nuclear materials at SRS, responded to the request because of the nature of the SNF and the capabilities of SRS, including the Savannah River National Laboratory. Germany funded the Savannah River National Laboratory to conduct research that would lead to a method to separate the fuel kernels from the graphite matrix, the first step in processing this fuel. DOE agreed to consider Germany's request for the following reasons: the spent nuclear fuel contains U.S.-origin HEU; success of the above-mentioned research on a laboratory scale; SRS expertise in nuclear engineering and the management of nuclear materials; and availability of hardened SRS facilities that could be used as is or modified to process and disposition this type of spent nuclear fuel.

A Statement of Intent between DOE for the United States; the Ministry of Education and Research for the Federal Republic of Germany; and the Ministry for Innovation, Science, and Research for the State of North Rhine-Westphalia (on behalf of the North Rhine-Westphalian State Government), was signed in late March and early April 2014. The Statement of Intent enabled DOE and the German signatories to continue evaluating the feasibility of this proposed project. DOE is conducting studies and reviews required to determine whether to proceed with acceptance of the spent nuclear fuel for processing and disposition, including preparation of this EA, and certain technical and engineering work.

Development efforts to date have demonstrated the feasibility of a vapor-digestion technology, extended the technology to the concurrent digestion of multiple unirradiated pebbles of spent nuclear fuel from Germany,), and reached the operation of an engineering-scale system (one-fifteenth scale) which integrates off-gas treatment components with the vapor digestion technology. Secondary process equipment and off gas treatment components are based primarily on previously-demonstrated technology. The next steps include a scale-up maturation process.

Scale-up maturation is a multi-year development program to address technical considerations related to the processing of the spent nuclear fuel from Germany. The maturation approach reviews the progress of development efforts to date, identifies technology needs and risks, prioritizes a plan for addressing those technology needs while mitigating risks, and considers available technology to accelerate technology development and deployment. Savannah River National Laboratory's efforts will principally focus on the removal of graphite and silica carbide from the fuel pebbles and kernels using the proposed vapor digestion process. The vapor digestion process will be integrated with feed preparation, off gas handling, and fuel disposition technologies. Operations will be demonstrated for implementation in a remote-handling facility.

Future development activities to advance the technology will involve several major maturation activities. These include remote opening and handling of the CASTOR casks, design of a fully-integrated prototypical digestion system, operation of prototypical equipment in a remote-handle configuration, and obtain critical process data using irradiated fuel kernels and individual pebbles. The maturation approach will also address essential safety, security, and facility interface issues which include facility permitting, waste disposal, and final fuel disposition. All of these research activities are being conducted under Categorical Exclusion B3.6 (Small-Scale Research and Development, Laboratory Operations, and Pilot Projects documented in a series of Categorical Exclusion Determinations prepared by the SRS NEPA Compliance Officer [DOE 2013f, 2013g, 2014f, 2015b, 2015c]). Should future research and development requirements be different from those evaluated and approved in these evaluations, additional NEPA reviews will be conducted prior to initiating those activities.

Savannah River Site Capabilities. The facilities and capabilities proposed for processing this spent nuclear fuel are unique to DOE and SRS. H-Canyon, which began operating in 1955, is the only hardened nuclear chemical separations plant still in operation in the United States. Historically, H-Canyon was used to recover uranium-235 and neptunium-237 from fuel irradiated onsite in nuclear production reactors and from domestic and foreign research reactor spent nuclear fuels. In 1992, however, the production reactors were shut down and DOE determined that recovery of HEU for nuclear weapons production was no longer necessary. In 2003, DOE's HEU disposition program began using H-Canyon to blend down HEU with natural uranium to make low-enriched uranium[12] (LEU) for use as a commercial nuclear reactor fuel for Tennessee Valley Authority nuclear power reactors (SRNS 2012a). In addition, H-Canyon continues to be used to separate and recover uranium from spent nuclear fuel and other highly radioactive materials for reuse and to prepare the residuals for disposal through the SRS Liquid Nuclear Waste Facilities. HEU is thereby rendered unusable for a nuclear weapon.

L-Area was initially constructed as a nuclear reactor for use as a nuclear material production facility in the 1950s. The reactor was permanently shut down in the 1980s, but the ancillary facilities have continued to support SRS missions, primarily receipt, storage and shipment of spent nuclear fuel and other special nuclear materials.

The Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Acceptance Program has been in operation since 1996, and is managed by the National Nuclear Security Administration. This Program is a vital part of current U.S. strategy to secure HEU and other fissile and radiological materials of U.S.-origin that may be attractive for non-peaceful purposes. The majority of spent fuel assemblies returned to the United States under the FRR SNF Acceptance Program are received and stored in L-Area.[13] The building was constructed to meet standards for nuclear material production and processing, and has maintained its structural integrity. In addition, in the early 2000s, research and development was conducted at SRS for the melt and dilute technology, a method for stabilizing spent nuclear fuel that is now proposed under the L-Area Alternative (see Chapter 2). During that timeframe, conceptual design for implementation of the melt and dilute technology in L-Area facilities was initiated but later halted. Aluminum-clad fuels stored in

[12] Low-enriched uranium has a concentration of the isotope uranium-235 above that of natural uranium (0.7 percent), but less than 20 percent.

[13] Through December 2014, approximately 51 shipments of material in 242 casks have been shipped to and received at U.S. ports under the FRR SNF Acceptance Program. Of those, 40 shipments containing 222 casks were received at Joint Base Charleston-Weapons Station then transported to SRS. All shipments occurred without incident. Materials from over 29 countries including Japan, Sweden, Germany, Slovenia, Romania, Turkey, South Africa, Australia, Chile, and Indonesia have been received at SRS. DOE would apply the experience and expertise gained from the FRR SNF Acceptance Program to this Proposed Action.

L-Area could also provide a potential source of the aluminum and uranium needed for the melt and dilute process.

The SRS Liquid Nuclear Waste Facilities are an extensive, integrated processing and disposition system comprising several facilities and technologies that do not exist elsewhere in the United States. The Liquid Nuclear Waste Facilities include storage, processing, and disposal facilities. The facilities include the H- and F-Area Tank Farms, high-level radioactive waste (HLW) storage tanks connected by an extensive piping system; the Defense Waste Processing Facility (DWPF); the Glass Waste Storage Buildings; and the Saltstone Production and Saltstone Disposal Facilities. Processed waste streams exiting H-Canyon are stored in the H-Area Tank Farm pending additional processing. The high-activity portion of the waste streams is processed into a vitrified waste form at DWPF. Canisters of the vitrified HLW are stored in the glass waste storage facilities pending permanent disposal in a repository. The low-activity portion of the waste streams is stabilized as a cementatious slurry in the Saltstone Production Facility then pumped into concrete Saltstone Disposal Facility vaults in Z-Area. As described in Chapter 2, these facilities are integral to processing the spent nuclear fuel from Germany for disposition.

1.3 PURPOSE AND NEED

DOE's purpose and need for the receipt, storage, processing, and disposition of the spent nuclear fuel from Germany is to support the U.S. policy objective to reduce, and eventually to eliminate, HEU from civil commerce (White House 1993). This action would further the U.S. HEU minimization objective by returning U.S.-origin HEU[14] from Germany to the United States for safe storage and disposition in a form no longer usable for an improvised nuclear device, a radiological dispersal device, or other radiological exposure device. Although HEU is the primary concern, the spent nuclear fuel contains other radioactive materials that could, in the wrong hands, be used to create radiological dispersal devices, commonly referred to as dirty bombs, that could cause harm to people and the environment.

1.4 PROPOSED ACTION

If the current feasibility studies show adequate promise, and DOE and Germany decide to proceed with the project, the German government would work with DOE to transport spent nuclear fuel in chartered ships across the Atlantic Ocean to Joint Base Charleston-Weapons Station, near Charleston, South Carolina. Joint Base Charleston-Weapons Station is a military installation with port facilities and security appropriate for accepting and handling such cargo. The spent fuel would be transported in casks that have been certified to meet international standards for Type B transportation packaging[15], and have a Certificate of Compliance from DOE and a Certificate of Competent Authority from the U.S. Department of Transportation as Type B casks for transport within the United States. From Joint Base Charleston-Weapons Station, the casks would be

[14] Prior to irradiation, the fuel contained approximately 900 kilograms (1,980 pounds) of HEU.

[15] Type B packages are required for the transport of highly radioactive material. Type B packages must withstand, without loss of contents, normal transport conditions such as heat, cold, vibration, changes in pressure, being dropped, compressed, sprayed with water, or struck by objects, as well as more serious accident conditions. These requirements are demonstrated during the licensing process for each Type B package through rigorous testing in accordance with 10 CFR Part 71.

transported to SRS on dedicated trains in accordance with applicable U.S. regulatory requirements. **Figure 1-1** shows the locations of facilities for the proposed activities.

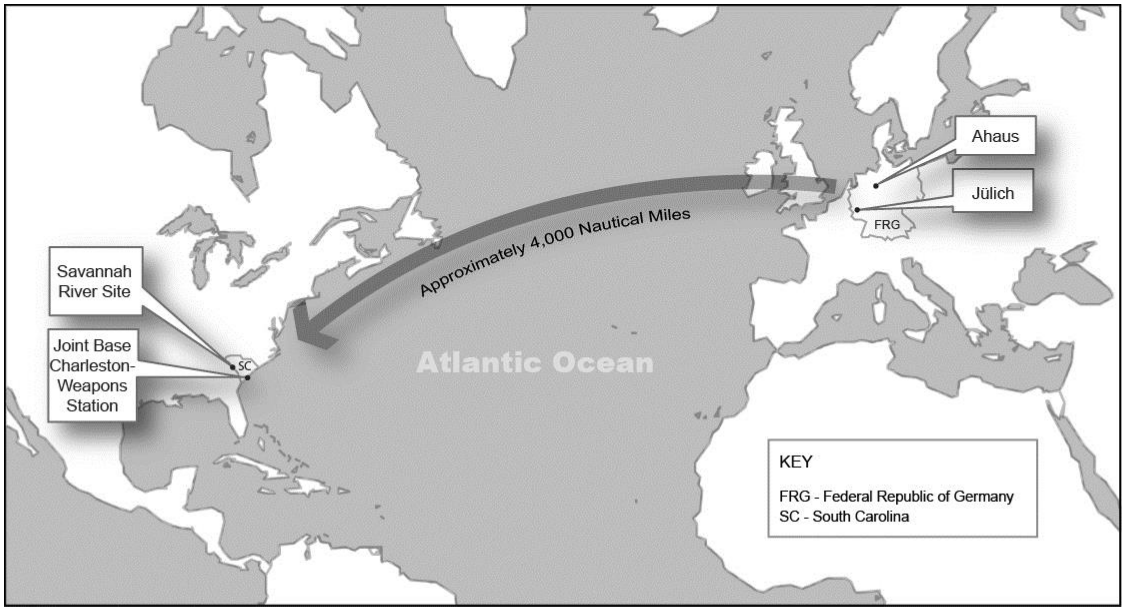


Figure 1-1: Proposed Project Locations

The spent fuel would be stored in CASTOR casks, the Type B transportation casks in which it would be shipped, on pads in H- or L-Areas, or both, until installation of the new carbon digestion equipment needed for initial processing of the spent nuclear fuel was completed. DOE would use the carbon digestion process, installed in either H-Area or L-Area, to separate the fuel kernels from the graphite matrix as the first step in preparing the spent fuel for disposition. Depending on the alternative, the following SRS infrastructure and facilities would be used in the process: E-Area; H-Area; H-Canyon; L-Area; and the Liquid Nuclear Waste Facilities, including the tank farms, DWPF, saltstone facilities, and glass waste storage facilities. Alternatives for implementing the Proposed Action, including the facilities required, are described in Chapter 2.

As specified in the Statement of Intent, any decision by the Participants (signatories to the Statement of Intent) to proceed with the transportation of the spent fuel for acceptance, processing, and disposition depends on compliance with all applicable requirements of United States law and DOE requirements, including NEPA, and resolution by the Participants of any technical, financial, and legal issues that may be identified during consideration of the feasibility of the project and development of an appropriate legal framework.

1.5 PUBLIC INVOLVEMENT

DOE announced its intent to prepare this EA with publication of a Notice of Intent in the Federal Register on June 4, 2014 (79 FR 32256). DOE invited Federal agencies, state and local governments, Native American tribes, industry, other organizations, and members of the public to submit comments on the proposed scope of the EA. The public scoping period opened with publication of the Notice of Intent, and closed on July 21, 2014. DOE has continued to accept and consider comments throughout preparation of this Draft EA. A public Scoping Meeting was held on June 24, 2014, at the North Augusta Community Center, North Augusta, South Carolina.

Approximately 227 public comment documents, including those in two letter campaigns, have been received since the public scoping period opened. Section 1.5.1 is a summary of those comments. Section 1.5.2 explains how the comments have been considered in preparing this Draft EA.

1.5.1 Summary of Comments

Comments in Support of the Proposed Project. Comments made in support of the proposed project include:

- The potential for socioeconomic benefits from the proposed action.
- The availability of unique facilities, infrastructure, and technical expertise at SRS.
- The SRS safety and environmental record.
- The contributions to nuclear nonproliferation that would be made by the proposed project.
- The perceived benefit that German funding of the proposed project could result in new technologies that could potentially be used later for treating domestic wastes, could secure jobs and retain technical expertise, and would defray the costs of operating and maintaining H-Canyon and other SRS infrastructure.
- The opportunity to utilize H-Area to down blend HEU with natural or depleted uranium to produce LEU suitable for reactor fuel for commercial use.

Comments in Opposition to the Proposed Project. Comments made in opposition to the proposed project include:

- Concern that accepting and processing the spent nuclear fuel would impact DOE's legal obligations, commitments, funding and schedule for treating and disposing SRS legacy wastes and permanently closing associated facilities such as the F- and H-Area Tank Farms and H-Canyon[16].
- Concern that import of the material could result in the import of spent fuel and other radioactive materials from other countries, turning SRS into a radioactive waste storage and disposal facility.
- Concern that more waste should not be brought into South Carolina because there is no disposition path out of SRS or South Carolina for the spent nuclear fuel or HLW that would result from processing.
- Concern about potential environmental and human health risks associated with processing the spent nuclear fuel.
- Concern that SRS facilities are becoming outdated; in particular that there have been operational problems at H-Canyon and that the facility is in need of substantial renovation and restoration. Commenters questioned whether these operational challenges at H-Canyon could affect the ability to process the spent nuclear fuel from Germany after it is at SRS.

[16] This comment is addressed under the heading *Content of the EA/Concerns about the Potential Impacts of the Proposed Project/Processing in Germany* in Section 1.5.2.

- Concern that export of the spent nuclear fuel from Germany may be prohibited under German or European Union law on the basis that the reactors in which the fuels were irradiated are power reactors (not research reactors).
- Concern that any LEU created by down blending of HEU to LEU for potential use as commercial reactor fuel would not meet the specifications for LEU power reactor fuel, particularly with regard to the nature and isotopic content of the resulting down blended product.

Project Funding. Commenters questioned whether the German commitment to fund the project would include long-term storage and eventual disposition. Commenters wanted proof of Germany's long-term financial commitment to the project, suggesting that a contract be signed detailing the financial arrangements before any spent fuel is shipped to the United States.

Relationship to Other Spent Graphite Fuels. Commenters referred to the possible similarities between the spent nuclear fuel from Germany and the spent nuclear fuel from the domestic Ft. St. Vrain and Peach Bottom reactors, which is in storage at other DOE facilities:

- Some commenters suggested that the Ft. St. Vrain and Peach Bottom fuels might be amenable to the same treatment that would be developed for the spent nuclear fuel from Germany.
- Other commenters wanted DOE to disclose in this EA any plans to move the Ft. St. Vrain or Peach Bottom fuels to SRS for processing, or wanted an Environmental Impact Statement (EIS) prepared if the scope of the program were to expand to include processing of these materials.

Need for an EIS. Commenters suggested that an EIS is more appropriate than an EA for this proposed project because of the duration, complexity, and perceived uncertainties about the project.

Content of the EA. Commenters requested that specific details about the Proposed Action and alternatives be described in this EA. For example, commenters requested full characterization of the radioactive materials and transport containers; number of casks and shipments; shipment schedule; details about current storage, preparatory activities and transportation in Germany; details about ocean transport and overland transport from the port to SRS; country of origin of the spent nuclear fuel and any other radioactive materials; and details about the processing and disposition at SRS.

Commenters also requested detailed analyses for all aspects of the Proposed Action and alternatives, including information on the potential human health risk associated with each aspect of the proposed project, including transportation, handling, processing, and disposal; evaluation of the economic and socioeconomic implications of the proposed project, including a detailed economic evaluation of the five counties surrounding SRS and the potential economic impacts on that area; impact of the proposed project on low income and minority populations; evaluation of the impact of potential emissions on air quality and greenhouse gases; and transportation and facility accidents.

Concerns about the Potential Impacts of the Proposed Project. Commenters expressed concerns about transportation safety and the potential impacts of a catastrophic transportation accident;

about seismic risk at SRS and potential impact on safety of operations, citing newly published studies suggesting a higher seismic risk than had previously been considered; about the potential effect of the local warm and humid climate on the ability to safely handle and store the containers, and the potential for more rapid transport of potential contaminants through the local ecosystem given the high water table and proximity to the Savannah River; and about the economic implications to the economy of the coastal region from a catastrophic event during ocean transit, off-loading at the port, and transport to SRS.

Processing in Germany. Some commenters stated that the spent nuclear fuel does not need to be imported because leaving it in Germany does not pose proliferation concerns. Commenters also suggested that rather than transporting the spent fuel to the United States for processing, DOE could provide the technology and if necessary, personnel, to Germany. Doing so, commenters suggested, would avoid impacts on the global commons and in the United States and not add to the growing inventory of HLW in the United States.

DOE Waste Management Policy and Strategy. Commenters expressed concern that the United States has failed to implement a long-term nuclear waste management strategy and cited issues related to the Mixed Oxide Fuel Program and the Waste Isolation Pilot Plant. Commenters suggested that in the absence of a long-term nuclear waste management strategy, the importation of the German waste would only add to the long-term storage burden of spent nuclear fuel, HLW and plutonium at SRS with no off-site disposition solution yet identified.

Public Opinion and Technical Papers. Commenters submitted editorials and opinion pieces from several local newspapers expressing opposition to the project on the basis of the concerns described previously in this section. Commenters also submitted technical papers, mostly in German, purporting to demonstrate that the AVR and THTR reactors are power reactors.

1.5.2 DOE Consideration of Comments

DOE reviewed and considered all public scoping comments it received. A number of the comments reflect policy concerns that are not within the scope of this EA, but rather affect whether the project can or should be implemented. Those out of scope comments, although included in Section 1.5.1, are not addressed in this EA. The remaining comments were addressed as follows.

Comments in Support of the Proposed Project. DOE notes all comments in support of the proposed project.

Comments in Opposition to the Proposed Project. DOE notes all comments in opposition to the proposed project. Specific concerns raised in opposition have been addressed in this EA as described in the remainder of this section and as follows:

- DOE added an alternative for processing the spent nuclear fuel using a melt and dilute technology in L-Area. This technology would provide an alternative to using H-Canyon, H-Area facilities, and DWPF, and would address concerns regarding availability of H-Canyon and impacts on schedule for treating and disposing SRS legacy wastes and permanently closing associated facilities such as the HLW Tank Farms and H-Canyon.
- As discussed in Section 2.3.3 of this EA, DOE dropped from consideration one of the options identified in the Notice of Intent, down blend of the HEU to LEU for reuse as

reactor fuel. As pointed out by commenters, the nature and isotopic content of the spent nuclear fuel from Germany makes LEU from down blending this HEU unsuitable for use in commercial reactor fuel.

- In a letter dated May 8, 2015 (Kraus 2015), the German Ministry for Education and Research, citing the *Report of the Federal Government of Germany for the Fifth Review Meeting of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management* submitted to the depository of the Joint Convention in October 2014 (BMUB 2014), stated that AVR and THTR were operated as experimental and demonstration reactors for the purpose of demonstrating the viability of the graphite pebble bed reactor technology, and that accordingly, these reactors are not classified as commercial nuclear reactors.

Project Funding. Project funding is outside the scope of this EA. DOE has received funds from Germany for the preliminary phase of this project, which includes preparation of this EA. DOE would not continue without assurances, through signed agreements, of continued funding.

Relationship to Other Spent Graphite Fuels/Need for an EIS. DOE has determined that an EA is the appropriate level of NEPA analysis for the Proposed Action. DOE does not have plans to expand the Proposed Action to include other fuels or materials. Following completion of the Final EA, DOE will either issue a finding of no significant impact (FONSI) or undertake an EIS, or cancel or modify the proposed project.

Content of the EA/Concerns about the Potential Impacts of the Proposed Project/Processing in Germany. The Proposed Action and alternatives are thoroughly described and analyzed in this Draft EA. Specifically, Chapter 1 explains the background for this proposed project, including the reactor type of the AVR and THTR. Chapter 2 of this Draft EA provides detailed descriptions of the alternatives proposed for implementing the Proposed Action; these details are needed to properly perform the impact analysis. The description of the Proposed Action includes descriptions of ocean transport, receipt and offloading at Joint Base Charleston-Weapons Station (the port of entry for the spent nuclear fuel from Germany), overland (rail) transport to SRS, and technical processing and disposition options proposed for SRS. Chapter 3 describes the environmental conditions relative to the impact analysis for each resource area, including seismic, climatological and socioeconomic characterizations. Chapter 4 details the analyses performed for this Draft EA and the potential impacts of the alternatives.

DOE conducted impact analyses for transportation, processing, and storage for both normal operations and postulated accidents using the most currently available seismic, environmental, and population data. These analyses show the potential impacts on human health of all aspects of the proposed project. The potential impacts on waste management capabilities, including storage at SRS pending permanent disposal, are also evaluated. Analyses for this Draft EA were performed commensurate with the potential level of impact. The potential impact of the Proposed Action on completion of cleanup of legacy waste and closure of facilities at SRS is discussed in Cumulative Impacts, Section 4.3.2.

Consistent with the requirements of Executive Order 12114, *Environmental Effects Abroad of Major Federal Actions*, evaluation of impacts on the global commons, in other words, on areas not within the sovereignty of any country, are evaluated. Activities occurring within other countries would be addressed by those countries in accordance with their requirements.

Providing the technology to Germany for processing rather than bringing the fuel to the United States for processing at SRS would be similar to the No Action Alternative described in Section 2.2.

1.6 LAWS, REGULATIONS, PERMITS AND CONSULTATIONS

Many Federal laws have been passed since the early 1960s to improve the quality of the environment by broadly addressing environmental media and industrial activities. The major laws (as amended) include: the Federal Water Pollution Control Act, the Clean Air Act of 1970, and the Solid Waste Disposal Act of 1965 as amended by the Resource Conservation and Recovery Act of 1976 and the Hazardous and Solid Waste Amendments of 1984. In addition, many other laws have been passed to protect more specific aspects of the natural or human environment. The Endangered Species Act of 1973, Migratory Bird Treaty Act of 1918, National Historic Preservation Act of 1966, and Native American Graves Protection and Repatriation Act of 1990 are a few of the more narrowly focused laws. Each law requires implementing regulations passed by agencies charged with enforcing those laws. The regulations and programs developed under these regulations apply to industrial and governmental activities, including those undertaken by DOE.

Laws such as the Occupational Safety and Health Act of 1970 are meant to ensure worker and workplace safety, including workplaces free from recognized hazards such as exposure to toxic chemicals, excessive noise levels, and mechanical dangers.

The Atomic Energy Act of 1954 (as amended), provides the basic statutory framework for DOE’s use and management of radioactive materials.

The National Environmental Policy Act of 1969 applies to Federal agencies and actions. This law established a national policy of environmental protection and directs all Federal agencies to utilize a systematic, interdisciplinary approach incorporating environmental values into decision making. NEPA requires that environmental information be made available to both decision makers and the public before decisions are made and actions taken.

Executive Orders issued by the President and applicable only to Federal agencies have the force of law and generally address a specific subject. Applicable to this proposed activity is Executive Order 12114, *Environmental Effects Abroad of Major Federal Actions*, which requires evaluation of major Federal actions significantly affecting the environment of the global commons outside the jurisdiction of any nation (e.g., the oceans or Antarctica).

Requirements applicable to the Proposed Action are discussed for each resource area in the respective sections of Chapter 3, Affected Environment, and Chapter 4, Impact Analysis, of this EA.

2 ALTERNATIVES

DOE is considering two action alternatives to implement the Proposed Action, as well as the No Action Alternative, as required by DOE's NEPA implementing procedures (10 CFR 1021.321(c)). The two action alternatives differ in processing technology and location at SRS where the processing would occur. The H-Area Alternative (so named because most activities would involve H-Area facilities) includes three processing options (Vitrification Option, Low-Enriched Uranium (LEU) Waste Option, and LEU/Thorium Waste Option) that use H-Canyon to differing extents; the L-Area Alternative (so named because the alternative would involve mostly L-Area facilities) would implement melt and dilute processing in L-Area. These action alternatives and the associated processing options are described in the following sections.

2.1 ACTION ALTERNATIVES FOR ACCEPTANCE AND DISPOSITION OF GERMAN GRAPHITE FUEL

2.1.1 Overview

Under the action alternatives, the spent nuclear fuel would be transported from Germany and processed at the Savannah River Site (SRS) for final disposition. Implementing this action would result in the return of U.S.-origin highly enriched uranium (HEU) material to the United States where its constituents would be processed and converted to proliferation-resistant waste forms.

Under each action alternative, 30 shipments would arrive at SRS over approximately 3.5 years. Each shipment would typically consist of eight railcars, with two casks per railcar, packaged in a standardized (International Organization for Standardization [ISO]) container. At SRS, the cask tie downs and impact limiters required for shipping would be removed and the cask would be upended to the vertical position and transferred to a storage pad. The form and composition of the nuclear material would require storage in a Property Protection Area where security would be provided by fencing, locks, and lighting.

DOE has identified process options (referred to as carbon digestion) for removing the graphite surrounding the spent fuel kernels and is evaluating them for implementation in SRS facilities. In addition to a molten salt digestion process currently under development, a vapor digestion process has been identified for process development and technical evaluation. As discussed in Section 2.3.1, these processes were selected for further development after considering other technologies.

DOE has evaluated a spectrum of options for processing the spent fuel kernels following carbon digestion (SRNL 2014a). Four were deemed the most feasible and have been carried forward for further process development and technical evaluation. Three would be deployed in H-Area; one would be installed in a modified wing of L-Area Material Storage Facility. The four options for processing the kernels after carbon digestion are:

H-Area Alternative Options

- **Vitrification Option –** Dissolution of the kernels in H-Canyon with direct transfer of the dissolver solution to the existing Liquid Nuclear Waste Facilities.
- **LEU Waste Option** – Dissolution of the kernels in H-Canyon followed by solvent extraction in H-Canyon for separation of the uranium. The uranium solution would be

down blended and grouted (i.e. solidified by mixing with cement) to meet acceptance criteria for disposal as low-level radioactive waste (LLW). Thorium, other actinides, and fission products would be processed through the Liquid Nuclear Waste Facilities.

- **LEU/Thorium Waste Option** – Dissolution of the kernels in H-Canyon followed by solvent extraction in H-Canyon for separation of the uranium and thorium. The uranium/thorium solution would be down blended and grouted (i.e. solidified by mixing with cement) to meet acceptance criteria for disposal as LLW. Other actinides and fission products would be processed through the Liquid Nuclear Waste Facilities.

L-Area Alternative Option

- **Melt and Dilute Option** – Down blending and conversion of the kernels to a uranium-aluminum alloy in a melt and dilute process in L-Area. The resulting ingots would be stored in concrete overpacks on a pad in L-Area. Unlike the H-Area processing methods, the kernels would not be dissolved prior to final processing. Therefore, the melt and dilute process would minimize the liquid waste stream transferred to the Liquid Nuclear Waste Facilities.

Figure 2-1 shows the two action alternatives. The preliminary processing steps (white boxes), from removing the pebbles from the casks through carbon digestion, but not the facilities in which the activities would occur, are the same for both the H-Area Alternative and the L-Area Alternative. After carbon digestion, the processing steps for the two alternatives diverge (shaded boxes). The H-Area and L-Area candidate facilities considered for processing have robust structural features, established perimeter security zones, and sufficient area for cask storage and staging or construction of new facilities, if needed.

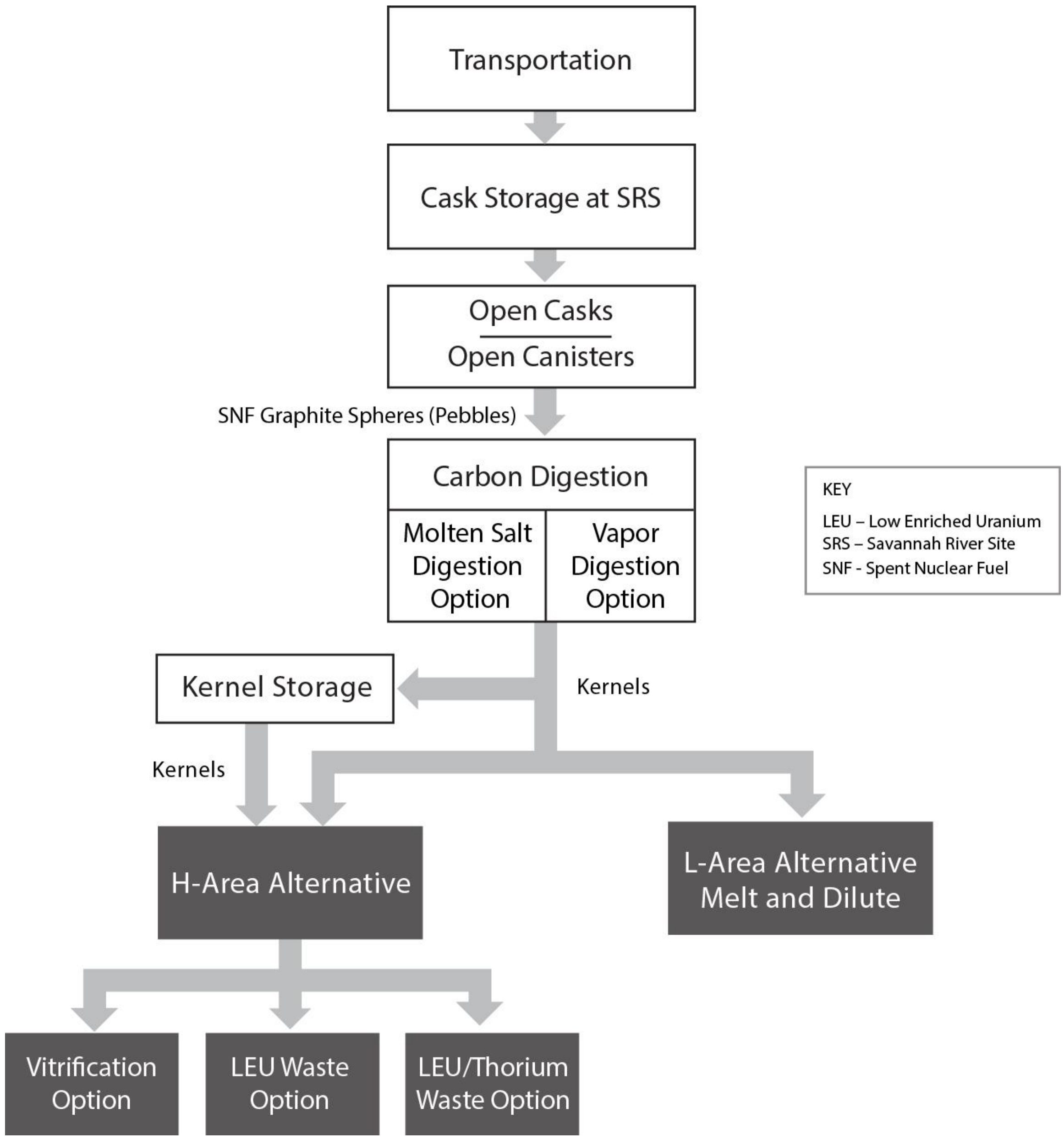


Figure 2-1: H-Area and L-Area Alternatives

2.1.2 Spent Nuclear Fuel and Packaging Characteristics

The spent nuclear fuel DOE is considering for acceptance and disposition consists of approximately 1 million graphite spheres or pebbles currently in storage in CASTOR casks at two locations in Germany. The number of pebbles in a cask varies, but on average there are about 2,200 pebbles per cask. As depicted in **Figure 2-2**, each pebble is approximately 60 millimeters (2.4 inches) in diameter and is composed of approximately 200 grams of graphite surrounding the fuel kernels; each sphere contains from 10,000 to 35,000 fuel kernels with varying quantities of uranium and thorium with uranium enrichments up to 81 percent. The fuel contained approximately 900 kilograms (1,980 pounds) of HEU prior to irradiation (Schütte 2012).

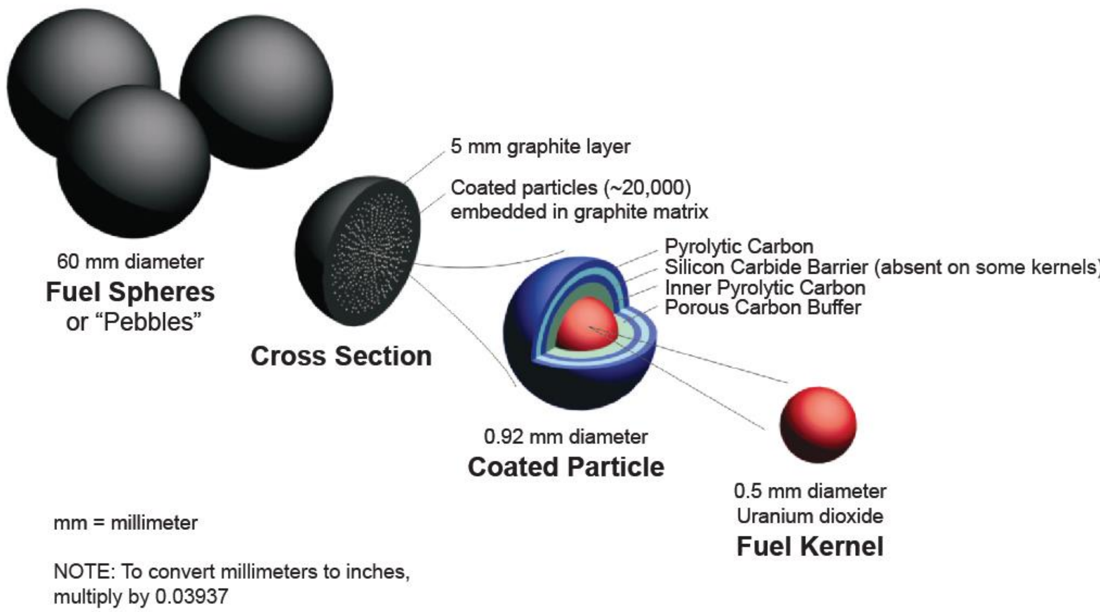
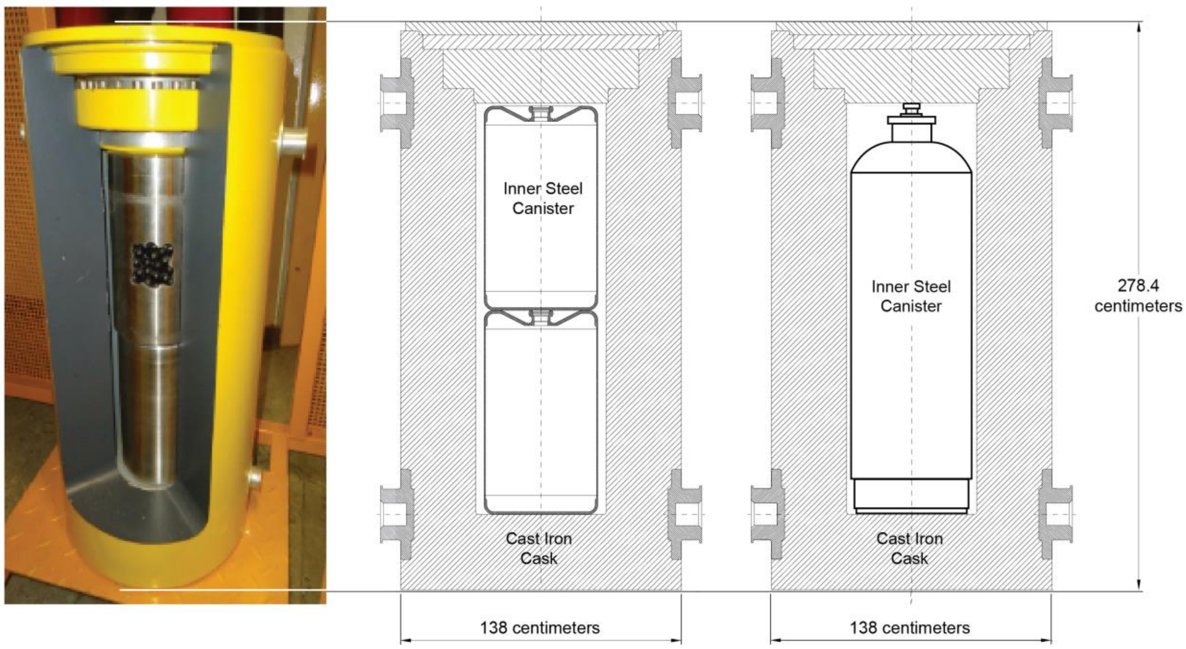


Figure 2-2: Composition of German Graphite Fuel

The pebbles are stored in 455 casks. **Figure 2-3** is a cutaway view of a CASTOR cask and interior schematics with the Arbeitsgemeinschaft Versuchsreaktor (AVR) and Thorium High Temperature Reactor-300 (THTR) canisters. Each cask is about 278.4 centimeters (109.6 inches) tall and 138 centimeters (54.3 inches) in diameter. The inside cavity of a cask is nominally 200 centimeters (78.7 inches) tall and 64 centimeters (25.2 inches) in diameter. The pebbles are contained in removable canisters inside the casks. There are 152 casks containing AVR fuel stored in Jülich, Germany, and 303 casks containing THTR fuel stored in Ahaus, Germany. The pebbles would remain in the CASTOR casks during transport of the spent nuclear fuel to the United States and while in storage at SRS pending processing.



Note: To convert centimeters to inches, multiply by 0.3937

Figure 2-3: Cutaway View of CASTOR Cask and Schematics with AVR and THTR Inner Canisters

2.1.3 Activities Common to H-Area and L-Area Alternatives

This section describes the activities common to both action alternatives. These activities include transportation from Germany to SRS, storage at SRS, and carbon digestion. Subsequent sections address the activities that are unique to the individual alternatives and options.

2.1.3.1 Transportation to SRS

The German government would be responsible for transporting the casks from the current storage locations to the United States. The transportation of the casks containing the AVR and THTR spent nuclear fuels would be conducted consistent with German laws and regulations until the casks become the responsibility of the United States. At Jülich and Ahaus, where the casks are in storage, the casks would be removed from their storage configuration, fitted with impact limiters on each end, and placed horizontally into ISO-standardized shipping containers. The German government would transport the shipping containers from the Jülich and Ahaus sites to a seaport in northern Germany where they would be secured aboard chartered ships certified to carry nuclear material. Consistent with Executive Order 12114, *Environmental Effects Abroad of Major*

Federal Actions, the environmental impacts analysis in this EA starts at the point of the transport ships entering the global commons.[17]

The ships would be certified to meet the requirements of the International Code for the Safe Carriage of Packaged Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes on Board Ships (INF Code).[18] Design and operational requirements for the three INF ship classes (with INF Class 1 being the lowest and INF Class 3 the highest) are addressed in a graded manner commensurate with the material being transported. Requirements address vessel stability after damage, fire protection, temperature control of cargo spaces, structural strength of deck areas and support arrangements, cargo securing arrangements, electrical supplies, radiological protection equipment, ship management, crew training, and emergency plans (WNTI 2007). In order to meet regulatory requirements for transporting the spent nuclear fuel, vessels used for transporting this material would, at a minimum, be INF Class 2 (DOE 2014a).

The shipping campaign from Germany would include about 30 shipments over an approximately 3.5-year period. Some shipments may include fewer shipping containers, but a nominal shipment would consist of 16 casks. To travel the roughly 4,000 nautical miles would require about 10.5 to 11.5 days; for purposes of analysis and to account for longer transit times due to weather or other events, DOE assumed a transit time of 15 days per shipment. The German government or its contractors would provide for physical protection of the shipment in Germany and the global commons and maintain physical protection responsibilities until transferred to the United States in U.S. territorial waters. Receipt and transfer of title and responsibility for the shipment, including security, would be consistent with the National Nuclear Security Administration's practices and protocols for foreign research reactor fuel receipts.

Members of the general public would not be exposed to radiation during transport to the United States. While at sea, some of the crew members would enter the hold and be in the vicinity of the shipping containers when performing inspections to ensure the cargo remains secure (that is, checking the tightness of the cargo tie downs). Inspections represent the largest potential for radiation exposure to crew members; inspections would be performed once per shift change (every 4 hours) and involve two crew members. The radiation dose received by these crew members would depend on the levels of radiation emitted from the shipping containers, the number and placement of the containers, the inspection durations, and the distance maintained from the containers during inspections. The external dose rate for a cask is about 1 millirem per hour at contact (DOE 2014a); the dose rate at the outer surface of the shipping container would be much lower.

Before entering the U.S. seaport, Joint Base Charleston-Weapons Station, vessels carrying the spent nuclear fuel would be in communication with appropriate personnel at the seaport to coordinate port entry and docking activities. Measures would be taken to ensure safety and security during the passage through the port entrance channel and in the Cooper River as the ship

[17] Global commons refers to areas that are outside the jurisdiction of any nation (e.g., the oceans or Antarctica).

[18] The INF Code is summarized at http://www.imo.org/OurWork/Safety/Cargoes/Pages/IrradiatedNuclearFuel.aspx .

travels to Joint Base Charleston-Weapons Station. A pilot may board the vessel to assist the passage to the designated wharf. Escort vessels or tugs may also assist the passage.

At Joint Base Charleston-Weapons Station, railcars for transport of the spent nuclear fuel would be staged in advance of the arrival of the ship at the dock. During the transfer of the cargo from the ship to railcars, security would be provided in accordance with a security plan. Authorized workers, assisted by ship crewmembers, would remove the tie-downs securing the shipping containers, attach rigging, lift the shipping containers using a crane, and place the containers on railcars where they would be secured for the trip to SRS. Each railcar would hold two shipping containers and there would be up to eight railcars per shipment.

The spent nuclear fuel would be transported by a commercial carrier using a dedicated train. The approximately 133-mile trip to SRS would take less than a day. National Nuclear Security Administration infrastructure and protocols for receipt of foreign research reactor fuel would be followed for these shipments, including Federal and State coordination protocols, and those for transport, security, and radiation control.

2.1.3.2 Cask Storage at SRS

Upon arrival at SRS, control of the train would be assumed by the SRS railroad group and an SRS locomotive would be used for onsite movement of the railcars. The casks containing the spent nuclear fuel would be removed from the shipping containers and stored on existing and/or new concrete or gravel storage pads in H-Area, L-Area, or a combination of the areas. Up to 40,000 square feet of storage capacity would be needed for the entire inventory of spent AVR and THTR fuel. The total area required would depend on the locations and configurations selected for storage. No modifications to the SRS site rail system are anticipated to support cask receipt and storage.

DOE would perform safety reviews, including for criticality safety, to confirm the safety of cask storage prior to receipt of the casks.

Upon receipt, the shipment would be subject to visual inspection, radiological survey, and data verification to ensure the casks meet all acceptance requirements. To remove the casks, the tops and sides of the shipping containers would be removed exposing the casks in the shipping frames. A mobile crane or equivalent would be used to transfer the casks from the railcars and to lift them into a vertical position. The crane would then place the individual casks on a transporter for transfer to the storage pad. Similar to the operation at the rail siding, after arriving at the storage pad a lifting apparatus would be connected to the casks and the crane would lift the casks from the transporter and place them into vertical storage positions. The casks would be placed with approximately 2 feet of spacing to allow for inspections.

After the casks have been placed on storage pads, they would be covered to protect them from the weather. Protection could be provided by covers for individual casks or by weather enclosures (steel super structure and fabric covers) that could be placed over an entire storage pad. The storage locations would be within property protection areas, with the necessary infrastructure (lighting, fencing, locks) to meet security requirements. Because the CASTOR casks are fabricated of metal and would remain sealed and covered while in storage, no additional features are expected to be necessary in the storage locations. While casks are in storage, inspections would be performed on a defined schedule. The casks would remain in storage until they were transferred for processing under one of the alternatives; the weather covers would be designed to provide protection for at least 10 years of cask storage. By properly sequencing the removal of casks from the storage pad for processing (that is, first in would be first out), none of the casks is expected to be in storage longer than 10 years.

Shipping frames would remain on the railcars and be returned via a commercial seaport for shipment back to Germany for reuse. A number of shipping frames would be retained at SRS for onsite movement of casks after shipments are completed.

2.1.3.2.1 <u>Storage in H-Area</u>

Casks storage in H-Area would be on existing storage pads and, if needed, on new pads built for this project. A portion of the casks could be stored on 4 existing concrete pads (approximately 20,000 square feet) in H-Area that would be made available by relocating equipment and equipment racks currently on the pads to other storage locations available in H-Area and F-Area. To accommodate all of the casks, an additional 14,000 square feet of storage capacity could be made available by constructing a new gravel or concrete pad and expanding the working area around an existing pad. In addition to the area of the additional storage capacity, approximately 10,000 square feet of land would be used during construction. Some improvements in H-Area, such as re-topping of existing roads, would be required if casks are stored in H-Area. All areas used for construction of storage pads, work areas, and roads would be within the existing H-Area.

2.1.3.2.2 <u>Storage in L-Area</u>

Storage capacity in L-Area would be provided by constructing a new storage pad. If all casks were to be stored in L-Area, a 40,000 square foot pad would be constructed of gravel or concrete. Gravel or asphalt roads circling the pads and connecting to existing roads would require construction of an additional 35,000 square feet of gravel or concrete surface, all within the existing L-Area.

2.1.3.3 Carbon Digestion of the Graphite Matrix

The initial step in processing the pebbles would be to separate the spent fuel kernels from the graphite matrix. The proposed process, carbon digestion, would be the same for both the H-Area and L-Area Alternatives. The purpose of the carbon digestion process would be to chemically separate the graphite coating from the kernels. Two carbon digestion technologies are being developed and evaluated by the Savannah River National Laboratory; one uses molten salt and the other a vapor to oxidize the graphite material that surrounds the spent fuel kernels. Both of these technologies are evaluated as options in this EA.

In preparation for carbon digestion, the casks would be moved from storage to the processing area in either H- or L-Area, depending on the alternative. Movement of casks would be similar to the methods used to place them into storage. A crane would lift the casks from their storage locations and a transporter and/or railcars would be used to transport them to H-Canyon or the modified L-Area Material Storage Facility. Using remote operations at the processing facility, the inner canisters would be removed from the casks, the lids would be cut off, and the pebbles of spent nuclear fuel would be emptied into a hopper. Pebbles would be transferred from the hoppers and fed into the digester in batches of approximately 500.

The carbon digestion processing options are described in the following two sections. Implementation of the carbon digestion processing options under the H-Area and L-Area Alternatives is described in Sections 2.1.4.1 and 2.1.5.1, respectively.

2.1.3.3.1 Molten Salt Digestion Option

Under the molten salt digestion processing option, the spent fuel pebbles would be loaded into a basket inside the reaction vessel (digester). Salt would be added to the digester and the temperature increased to about 600 degrees Celsius; the molten salt would digest the carbon shell and graphite matrix of the pebble, exposing the kernels. Some of the kernels have a silicon carbide layer that would not be digested by the molten salt. A caustic would be added to digest this layer and fully expose the kernels. As the carbon is digested, the exposed kernels would exit the basket and settle in an annulus in the bottom of the digester vessel; the size and shape of the annulus would keep the kernels in a criticality-safe geometry. After the digestion is complete (approximately one day), the molten salt would be drawn from the digester into a storage tank, leaving the kernels to be recovered from the bottom of the digester. The kernels remaining following digestion would be about 2 percent of the volume of pebbles fed into the digester.

The kernels, along with a small amount of salt, would be drained from the digester into a carbon steel can. Each can would be about 5 inches in diameter and 3 feet long and would hold kernels from two batches of pebbles (about 1,000 pebbles). Once filled, the can would be closed with a carbon steel lid. The can would be assayed to measure its radionuclide content and then moved either to storage or directly to processing. Can storage would be in a hot cell in H-Canyon, for the H-Area Alternative. Under the L-Area Alternative, the separated kernels would be processed to a final form rather than stored.

Salt used in the carbon digestion process would be regenerated for reuse. Regeneration would include the addition of an acid that releases the carbon in the molten salt as carbon dioxide. When returned to the digester, the regenerated salt would be augmented with new salt to make up for the quantity drained with the kernels and discharged in the off-gas. After a number of batches, the salt would no longer be able to be regenerated. Salt that can no longer be regenerated would be dissolved and transferred to the tank farms for processing at the saltstone facilities.

Off-gas from the digester and the salt regeneration process would be treated to remove cesium, strontium, actinides, and entrained particulates. These materials would be processed and disposed along with similar SRS wastes in the liquid and solid waste management systems.

2.1.3.3.2 Vapor Digestion Option

Under the vapor digestion option, spent fuel pebbles would be loaded directly into the digester. The digester temperature would be raised to its reaction temperature, between 600 and 700 degrees Celsius, and a gaseous oxidant (vapor) would be passed through the digester. The vapor would digest the carbon shell and graphite matrix of the fuel pebbles, converting the carbon to carbon dioxide and liberating the kernels. The gases and particulates from pebble digestion would be drawn from the digester and processed through an off-gas treatment system.

The liberated kernels would be removed from the digester for a polishing step and molten salt bath digestion. The quantity of salt used for digestion of the residual carbon on the kernels would be much smaller than that used in the molten salt digestion process. Some of the kernels have a silicon carbide layer that would not be digested by vapor. A caustic would be added to the molten salt bath to digest this layer and fully expose the kernels. As the residual carbon is digested, the exposed kernels would be collected in a vessel sized and shaped to maintain a criticality-safe geometry. After the residual carbon digestion of the kernels is complete (approximately one day), the molten salt would be drawn from the molten salt digester into a storage container. The kernels remaining following digestion would be about 2 percent of the volume of pebbles fed into the digester.

The kernels, along with a small amount of salt, would be drained from the digester and managed the same as they would be if the molten salt digestion process were used. They would be containerized, assayed to measure their radionuclide content, and then moved either to storage (H-Area Alternative) or directly to processing (L-Area Alternative). As with the molten salt digestion process, salt used in the carbon digestion process would be regenerated for reuse, and when the salt can no longer be used, it would be dissolved and transferred to the tanks farms for processing at the saltstone facilities.

Off-gas from the digester and the salt regeneration process would be treated to remove cesium, strontium, actinides, and entrained particulates. These materials would be processed and disposed along with similar SRS wastes in the liquid and solid waste management systems.

2.1.4 H-Area Alternative

Under this alternative, the spent fuel pebbles would undergo carbon digestion either by molten salt or vapor digestion as described in Section 2.1.3.3. The extracted uranium or uranium/thorium kernels would be dissolved in the H-Canyon dissolver, and then processed by one of three options. Under the Vitrification Option, the entire dissolver solution would be transferred to the Liquid Nuclear Waste Facilities for disposition as vitrified high-level radioactive waste (HLW) glass (the high-activity fraction) and LLW saltstone[19] (the low-activity fraction). Under the LEU Waste Option, the dissolver solution would be processed through H-Canyon to separate uranium from the rest of the solution so the uranium could be down blended, solidified, and disposed of as LLW. The LEU/Thorium Waste Option is similar to the LEU Waste Option, except that thorium would be removed along with the uranium for down blending, solidification and disposal as LLW.

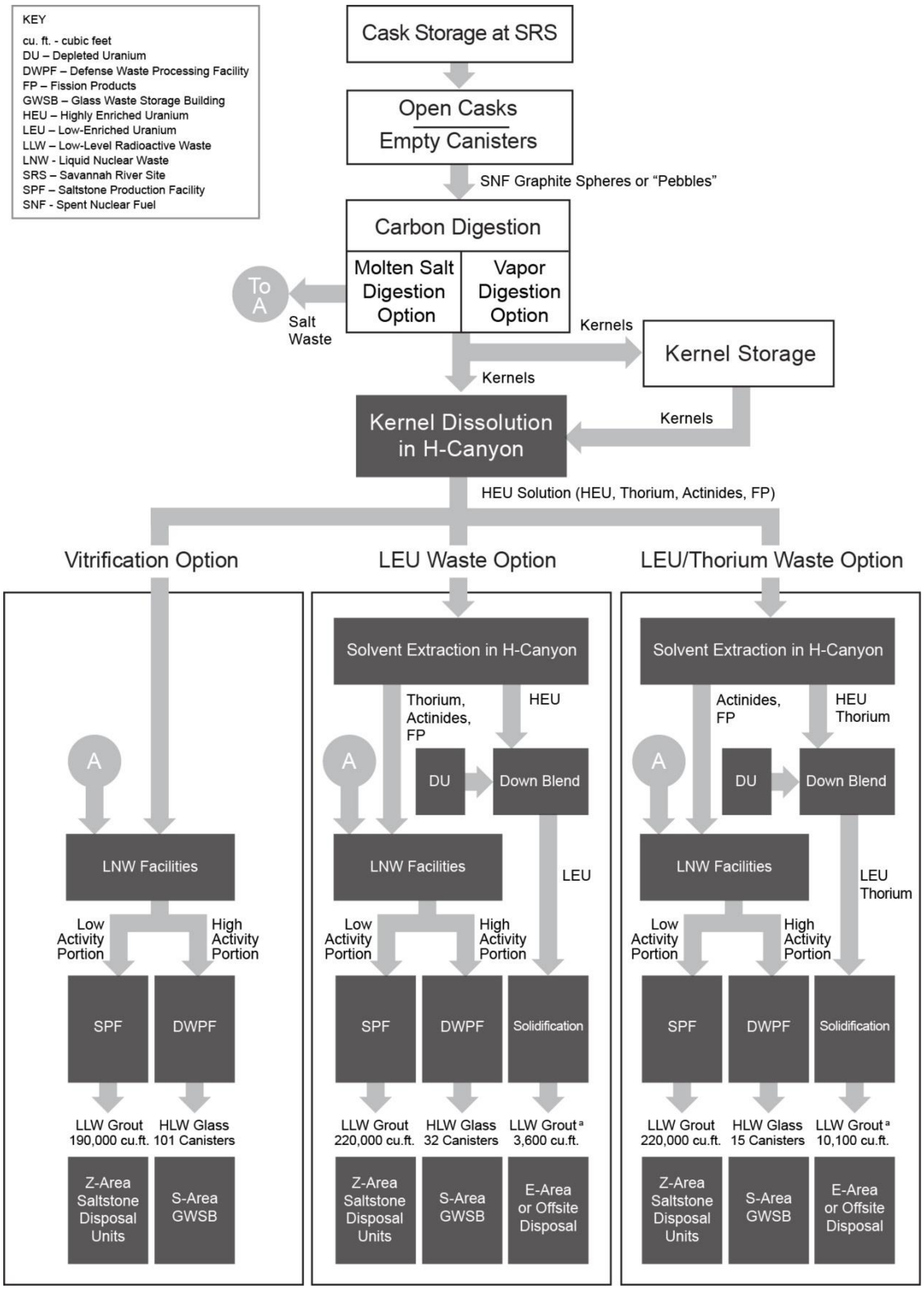
Figure 2-4 presents the H-Area Alternative and depicts the three options for processing the dissolver solution through H-Canyon, the Vitrification Option, the LEU Waste Option and the LEU/Thorium Waste Option. This figure shows the principal waste streams generated under each of the three options. Additional wastes, such as casks, canisters, and job control waste are not shown in Figure 2-4, but are identified in the process option descriptions and evaluated in the impacts analysis. Timelines showing the sequence and estimated durations of activities of the H-Area Alternative options are presented in **Figure 2-5** for the Vitrification Option and **Figure 2-6** for the LEU Waste and LEU/Thorium Waste Options.

[19] Saltstone is a concrete waste form created by mixing the low-radioactivity fraction of high-level radioactive waste with cement, ash, and slag.

Activities would occur on the "hot" side of H-Canyon and be performed remotely by operators working in shielded areas. Under all three options, H-Canyon would be modified to accommodate receipt of the spent fuel pebbles and to install the carbon digestion capability. A mobile cask platform would be installed in the Railroad Tunnel to allow access to the casks for lid removal. The Hot Shop would be modified with the installation of a canister staging rack, radiation monitors, equipment for cutting off the tops of canisters and inverting them, and a hopper for receiving pebbles.

H-Canyon Section 5 would be modified to accommodate the carbon digestion equipment. Existing equipment (a resin digestion tank, a waste tank, and dissolver) would be relocated. New equipment would be installed for two digester systems, including feed hoppers, digester vessels, remote manipulators, product-can turntables, and a salt transfer system. New ventilation equipment would be installed for the digestion process, including a high-efficiency mist eliminator, an off-gas condenser, and a condensate collection tank.

The H-Canyon bundle storage area (H-Canyon Section 3) would be modified with the installation of a rack for storage of kernel cans from the carbon digestion process. Modifications would be made to the Storage Pool to accommodate equipment for performing routine maintenance on the carbon digestion process equipment and for decontamination of failed equipment prior to disposal. Equipment would be fabricated, assembled, and tested prior to installation in H-Canyon. Other routine modifications such as installation of a canister grapple system on the hot canyon crane, piping changes, electrical and instrument changes would also be required.



[a] Values represent the as-generated volumes of one of the principle wastes from this option. As discussed in text, the as-disposed volume would be larger.

Figure 2-4: H-Area Alternative

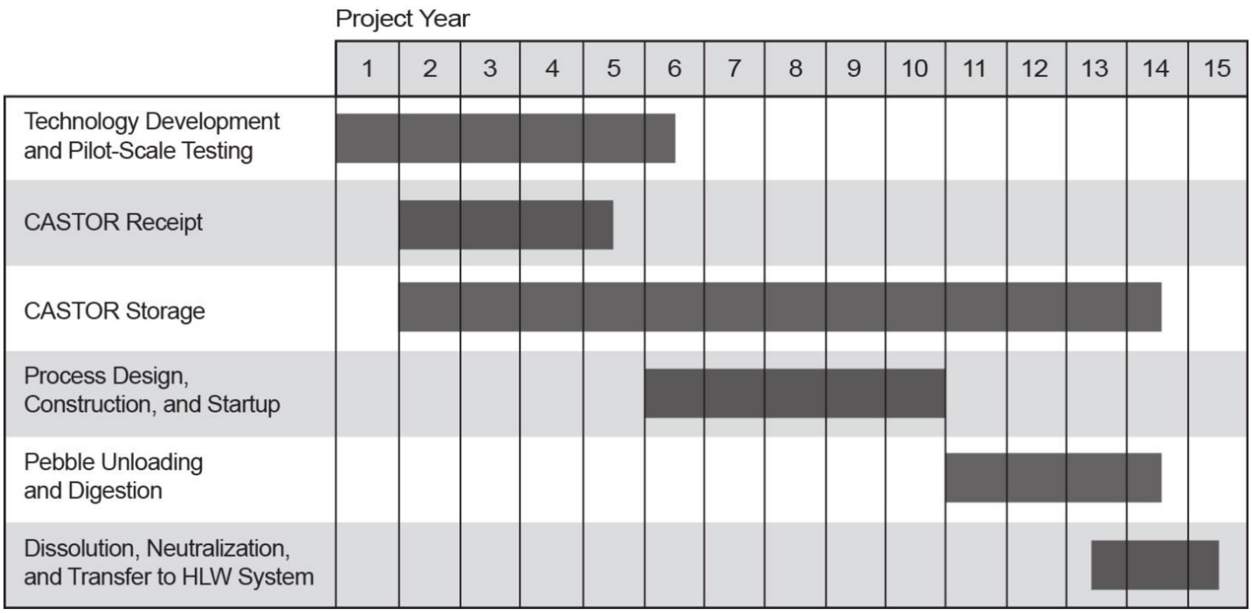


Figure 2-5: Vitrification Option Estimated Timeline

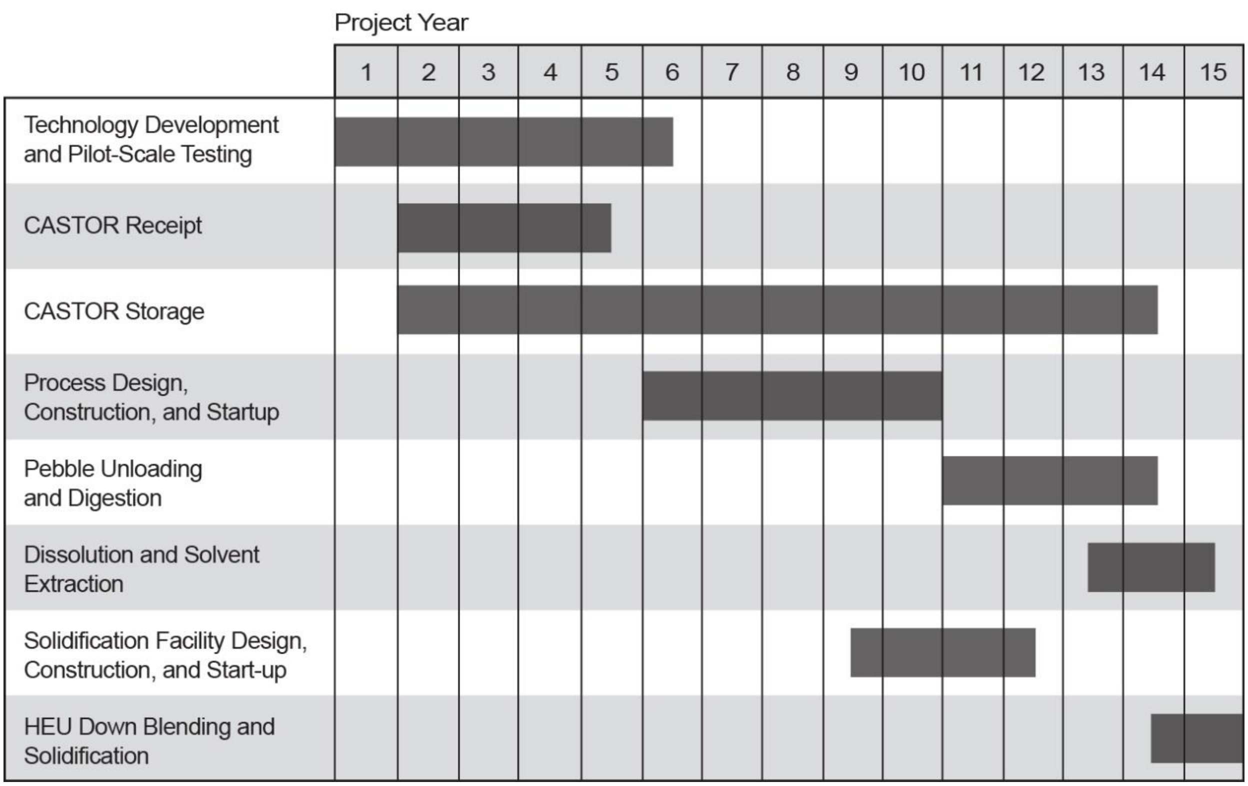


Figure 2-6: LEU Waste and LEU/Thorium Waste Options Estimated Timeline

Process equipment in a section of H-Canyon would be relocated to another section of the canyon or removed to make room for the new equipment. Removed equipment would be decontaminated as necessary prior to disposal in the E-Area facilities. The relocation and removal of equipment would not affect existing or planned H-Canyon operations.

2.1.4.1 Carbon Digestion under the H-Area Alternative

The first step in preparing the pebbles for processing would be removing them from the CASTOR casks. Casks would be transported from their storage locations to the H-Canyon Railroad Tunnel. In the tunnel, the casks would be opened and the inner canisters removed and transferred to a staging rack in the Hot Shop using the canyon hot crane. In the Hot Shop, through remote operations using the canyon hot crane, a canister would be placed into a cradle and its top cut off. The canister would then be inverted to dump the pebbles into a hopper. Pebbles would be moved in buckets from the hopper in batches of approximately 500 for placement in a digester. Two digesters, each with a design processing rate of 500 pebbles per day, would be used to process the pebbles. Accounting for less than optimal loading (for example, 950 rather than 1,000 pebbles processed per day between the two digesters) and a 75 percent operating efficiency, it is estimated that digestion of all of the pebbles would take approximately 3.5 years.

The kernels from the digestion process would be collected in cans as described in Section 2.1.3.3. Closed cans would be transferred to racks in the bundle storage area where they would be stored until they could be processed. Kernel storage cans would be dissolvable and sized to fit in the H-Canyon dissolver.

Salt waste from the carbon digestion process would be dissolved and transferred through existing H-Canyon piping to the tank farms for processing through the Liquid Nuclear Waste Facilities for disposal. The empty inner canisters would either be disposed by themselves as LLW or placed back into the CASTOR casks for disposal as LLW.

2.1.4.2 Uranium Kernel Dissolution under the H-Area Alternative

The cans containing the spent nuclear fuel kernels extracted in the carbon digestion stage would be moved using the canyon hot crane to an H-Canyon dissolver. Up to three cans of kernels would be placed in a dissolver containing a strong acid solution. Process development currently underway will determine operating parameters, including the amount of time required to dissolve the cans of kernels. Cans of kernels would continue to be added to the dissolver until the uranium concentration specification is reached, at which point the solution would be transferred for processing by one of the three options described in Section 2.1.4.3.

2.1.4.3 H-Canyon Processing and Disposition Options

Figure 2-4 depicts the three options for processing the dissolver solution through H-Canyon, the Vitrification Option, the LEU Waste Option and the LEU/Thorium Waste Option.

2.1.4.3.1 Vitrification Option

Under the Vitrification Option, the dissolver solution containing uranium, thorium, actinides, and fission products would be processed through the existing SRS Liquid Nuclear Waste Facilities to HLW glass and LLW saltstone waste forms for disposal.

The dissolver solution would be transferred to H-Canyon tanks where chemical adjustments would be made to meet tank farm acceptance criteria. Manganese would be added as a neutron poison (for criticality control). The waste would then be neutralized and transferred to the tank farm. The salt waste from the carbon digestion process (described in Section 2.1.3.3), containing up to 12 percent of the uranium and residual quantities of minor actinides, would also be routed to the tank farms.

The waste transferred from H-Canyon would be pretreated in the tank farms and result in two principal waste streams. The pretreatment would produce a high-activity stream containing the uranium, thorium, actinides, and most of the fission products. This stream would be routed to the Defense Waste Processing Facility (DWPF) where it would be combined with other materials and vitrified (that is, melted into a glass waste form). The molten glass would be poured into HLW canisters where it would cool and solidify. Sealed canisters of HLW glass would be transferred from DWPF to the glass waste storage facilities in S-Area for storage along with canisters of SRS HLW for eventual disposal in a HLW repository. The spent nuclear fuel from Germany would result in an estimated 101 canisters of HLW glass (SRNL 2014a).

The low-activity stream from pretreatment would contain relatively small quantities of the uranium, thorium, actinides, and fission products. This waste stream would be mixed with a grout in the SRS Saltstone Production Facility and disposed of in saltstone disposal units in Z-Area. Under this option, processing of the spent nuclear fuel from Germany would generate approximately 190,000 cubic feet of saltstone (SRNL 2014a).

No construction or major equipment modifications would be required for the Vitrification Option.

2.1.4.3.2 LEU Waste Option

Under this option, uranium would be separated from the dissolver solution containing uranium, thorium, other actinides, and fission products, down blended to LEU, and solidified into a LLW form for disposal at the SRS E-Area, offsite at DOE's Nevada National Security Site (NNSS), or offsite at a commercial LLW disposal facility. The balance of the process stream would be transferred to the existing SRS Liquid Nuclear Waste Facilities for processing to HLW glass and LLW saltstone.

Implementation of this option would require a new facility (uranium solidification facility) to house the solidification (cementation) process. A facility of 10,000 to 12,000 square feet would be constructed in the H-Area limited area (that is, on previously disturbed land inside the security fence); about half the space would be for the cementation operations and an equal amount of space would be for ancillary equipment (e.g., feed tanks, exhaust system). The cementation system would include a caustic supply system with a 1,000 gallon supply tank with agitator, two caustic supply pumps, and two caustic metering pumps; a mixing system with two cement head tanks and an agitated 600-gallon uranium solution feed tank; and two cementation stations with conveyors for positioning containers to be filled with grout. The uranium solidification facility would also have a conveyor system for moving filled containers to a lag storage area and a decontamination station in the event that a container needed to be cleaned prior to leaving the facility. Cementation stations would have local high-efficiency particulate air filters and would be connected to a facility ventilation system that would provide for local air treatment with a condenser, filtration, and fans. The facility ventilation system would be connected to the existing 292-H exhaust system for discharge through the H-Canyon stack.

Under the LEU Waste Option, the dissolver solution would be processed through the H-Canyon solvent extraction process. The solvent extraction process would produce two aqueous streams, one containing uranium and the other containing thorium, other actinides, and fission products. The aqueous uranium solution would be transferred to existing down-blending tanks adjacent to H-Canyon where it would be mixed with depleted or natural uranium to yield an LEU solution in which the fissile uranium content (uranium-233 plus uranium-235) would be reduced to acceptable levels for disposal. Depleted or natural uranium would be supplied by onsite inventories of uranium oxide or by uranium solutions from other DOE sites. The LEU solution would be transferred for short-term storage in an H-Canyon tank prior to solidification. The high uranium-232 content of the separated uranium would result in the ingrowth of short-lived daughter radionuclides that would emit energetic and penetrating gamma radiation. In order to keep radiation doses to workers as low as reasonably achievable (ALARA) the LEU solution would be promptly solidified.

The LEU solution would be stored in tanks in H-Canyon until transferred to the new uranium solidification facility for processing into a solid LLW grout. Caustic solution would be added to adjust the pH of the LEU solution; the resulting solution would be held in a feed tank equipped with agitators to keep material in suspension. Specially fabricated waste containers, sized to fit inside a CASTOR cask would be preloaded with disposable agitators. A container would be positioned in the cementation station. The liquid LEU would be metered into the waste container and a conveyor system would deposit dry materials into the container. Using the disposable agitator, the container contents would be mixed to assure uniform wetting of the dry material and distribution of uranium solution. After decoupling the agitator, the container would be moved by conveyor to a staging area for curing. Following a minimum 24-hour curing time, the containers would be capped and if necessary, decontaminated. To avoid unnecessary personnel exposure the closed containers would be placed in the CASTOR casks or otherwise shielded as soon as possible after the grout has cured.

The remaining aqueous stream from the solvent extraction process containing the thorium, other actinides, fission products, and the salt waste stream from the carbon digestion process would be neutralized and transferred to the Liquid Nuclear Waste Facilities. This waste stream would be processed as described under the Vitrification Option into HLW glass and LLW saltstone.

Under this option, the principal wastes would be an estimated 32 canisters of HLW glass produced in DWPF, approximately 220,000 cubic feet of LLW saltstone (SRNL 2014a), and approximately 3,600 cubic feet of grouted LEU LLW (Dyer 2015). The grouted LEU waste form would be poured into containers that would be placed into the casks for disposal. To meet disposal site requirements for fissile material content, an administrative limit of 900 grams of fissile material per cask would be imposed, meaning that only a portion of the capacity of each cask would be used; the balance of the cask would be filled with clean grout. The 451 CASTOR casks would be used for this waste, resulting in a LLW disposal volume of about 67,000 cubic feet (Dyer 2015).

2.1.4.3.3 LEU/Thorium Waste Option

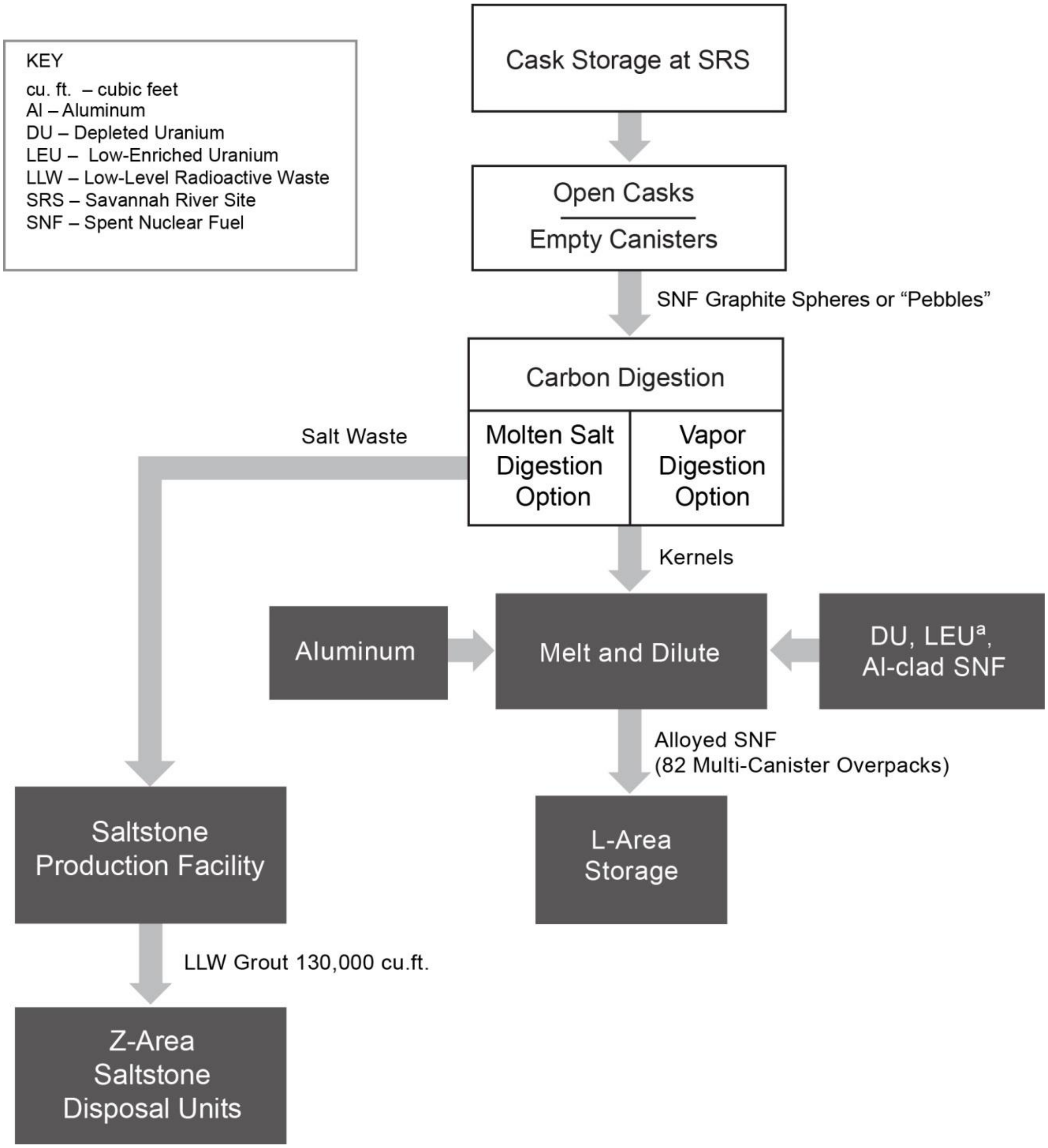
The LEU/Thorium Waste Option is the same as the LEU Waste Option, except that the thorium would be included in the aqueous waste streams with the uranium that would be down blended to LEU and solidified into LLW for disposal. Extracting both uranium and thorium would be accomplished by adjustments in the H-Canyon solvent extraction process and would not require any equipment or construction different than that under the LEU Waste Option.

The primary difference between this option and the LEU Waste Option would be in the volumes of the principal waste streams produced. Under this option, the principal wastes would be an estimated 15 canisters of HLW glass produced in DWPF, approximately 220,000 cubic feet of LLW saltstone (SRNL 2014a), and approximately 10,100 cubic feet of grouted LEU/thorium LLW (Dyer 2015). The grouted LEU/thorium waste form would be poured into containers. As many containers as possible would be placed into CASTOR casks, but there would be many more containers than the casks could accommodate. The CASTOR casks with containers of grouted LEU/thorium waste would represent a LLW disposal volume of approximately 67,000 cubic feet. Containers in excess of those placed in the CASTOR casks would account for an additional 5,500 cubic feet of grouted LLW requiring disposal (Dyer 2015). The disposal volume of this remaining grouted LLW would depend on disposal facility requirements. The volume would range from 5,500 cubic feet to tens of thousands of cubic feet, depending on whether additional packaging would be needed to meet disposal criteria. Early coordination with the disposal facility operator to determine packaging and disposal requirements would ensure that there would not be any delay in disposal of this waste.

2.1.5 L-Area Alternative

Figure 2-7 depicts the L-Area Alternative. Under this alternative, the spent fuel pebbles would undergo carbon digestion either by molten salt or vapor processing as described in Section 2.1.3.3. The extracted uranium and uranium/thorium kernels would then be converted into metal ingots through the melt and dilute process. The kernels would be blended with other uranium (if required to satisfy safeguards requirements) and combined with aluminum metal at high temperatures to produce an alloy. The alloy would be cast into ingots (approximately 4.2 inches in diameter and 47 inches long) that would be loaded into multi-canister overpacks (MCOs) that would be welded closed and placed on the L-Area pads in storage casks. A timeline showing the sequence and estimated durations of activities of the L-Area Alternative is presented in **Figure 2-8**.

Activities would be performed remotely within shielded hot cells. Areas of the L-Area Material Storage Facility would be modified to accommodate the activities and equipment for both the carbon digestion and melt and dilute processes. Both the new carbon digestion (either the molten salt or vapor digestion process) and melt and dilute processes would be installed in the Purification Wing of the L-Area Material Storage Facility (Purification Hot Cell Area). This area would require modification to accommodate the new equipment and processes. Modifications would take place within or adjacent to the existing structure. The two existing hot cells would be converted into four hot cells: an unloading cell, a digester and salt wash cell, an off-gas and solution handling cell, and an alloying furnace cell. These modifications would require removing the floor and piping in the two existing hot cells to create a cell space equivalent in height to that in H-Canyon and installing new walls to create the two additional cells. A new shielded dry transfer system would be installed to remove the cans of pebbles from the casks and move them on a dolly from the Stack Crane Area to the Purification Hot Cell Area. This system would also be used to move the ingots from the can-out area of the alloying furnace cell back to the Stack Crane Area for loading into MCOs. A new shielded truck bay with access to the facility through an airlock and shield door, with an associated shielded area for staging and removing waste would also be built. Upgrades would also be required to the heating and ventilation system: a sand filter, fans and stack to exhaust the process cell would be installed. Equipment would be fabricated, assembled, and tested prior to installation in L-Area Material Storage Facility.



[a] LEU that is not suitable for other purposes.

Figure 2-7: L-Area Alternative

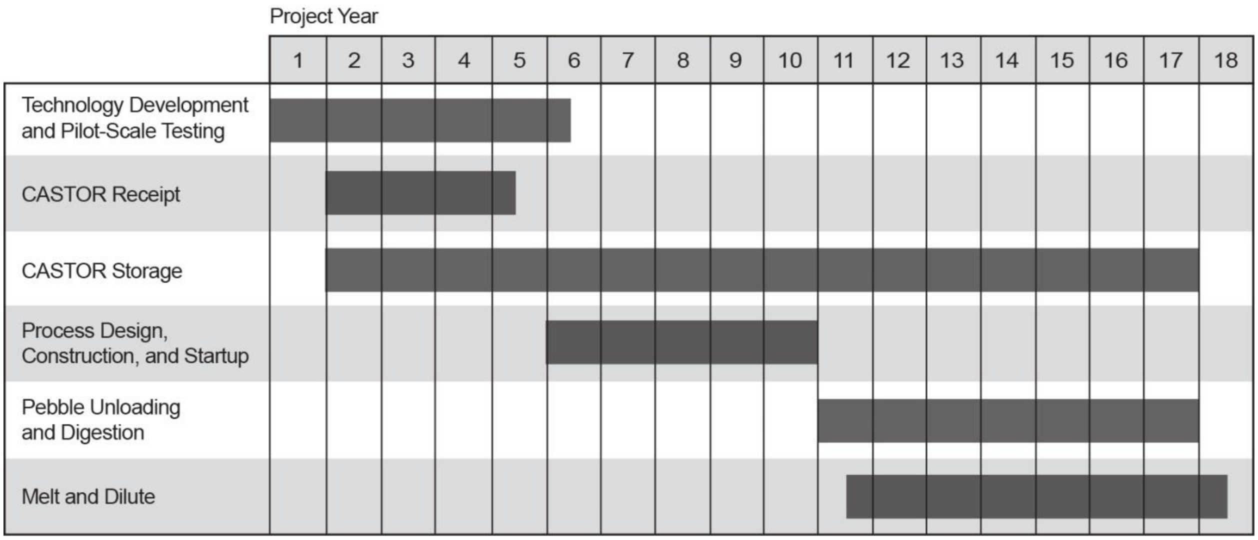


Figure 2-8: L-Area Alternative Estimated Timeline

2.1.5.1 Carbon Digestion under the L-Area Alternative

The first step in preparing the pebbles for processing would be removing them from the CASTOR casks. The CASTOR casks would be brought into the Stack Area of the L-Area Material Storage Facility from their storage location by a transporter and unloaded using the existing Stack Area crane. The inner canisters would be removed from the casks within the dry transfer system and moved on a dolly to the new process cell in the Purification Wing.

After being transferred to the unloading cell, the canisters would be assayed, their tops cut off and the canisters inverted to pour the pebbles into a hopper. Lag storage would be provided for staging pebbles for carbon digestion batch processing.

The pebbles would be metered from the hopper into a bucket for charging into the digester, where the kernels would be separated from the carbon matrix (see Section 2.1.3.3). Because of space considerations, only one digester would be installed in L-Area. Therefore, only half as many pebbles, approximately 500 pebbles per day, would be processed. Processing all of the pebbles would take twice as long as through H-Canyon, up to approximately 7 years. The separated kernels would not be stored, but would be transferred directly to the alloy furnace for processing. The used salt from the digester would be treated in the off-gas and solution handling cell and returned to the digester for reuse or managed in the saltstone facilities and disposed primarily as saltstone, a LLW, when no longer capable of being reused.

2.1.5.2 Uranium Kernel Processing and Disposition under the L-Area Alternative

The spent nuclear fuel kernels would be mixed with depleted uranium or LEU in the alloying furnace to dilute the isotopic concentrations of uranium-233 and uranium-235 to an acceptable level. Aluminum metal would then be added to the furnace to form an alloy with the uranium and thorium. Depleted uranium would come from SRS, the Paducah Site, the Portsmouth Site, or the Hanford Site. Low-enriched uranium that has been produced, but is not suitable for other purposes, could be used in the process; sources of such material are not known at this time. It is also possible that aluminum-clad SNF from L-Area at SRS or from the Idaho National Laboratory could be used to supply some of the aluminum and uranium needed to make the ingots.

The resulting aluminum-uranium-thorium ingots would be cooled and remotely moved from the breakout station in the furnace hot cell down a chute to a can-out capability in an adjacent lower level of the building. A transfer device would be used to transfer each ingot into a tight-fitting aluminum containment sleeve, which would then be remotely moved into a storage basket in a shielded transfer device mounted on a dolly. Once the storage basket is filled, the shield lid would be placed on the transfer device and the dolly would be moved by dumbwaiter to the main level, then back to the Stack Area.

In the Stack Area, the overhead crane would be used to unload the basket from the transfer device and load it into an MCO for storage. Each MCO would hold 28 ingots in 2 layers, each layer comprising a basket of 14 ingots. The MCO would be sealed, tested for leaks, then moved in a shielded transfer cask to the L-Area storage pad (originally constructed for CASTOR cask storage) where it would be loaded into a concrete storage overpack. Up to 5 MCOs, each 2 feet in diameter by almost 14 feet long, would be stored in each concrete overpack. The L-Area Alternative is projected to generate 82 MCOs and 130,000 cubic feet of saltstone. The MCOs would remain in storage pending a long-term solution for management of DOE HLW and SNF. The saltstone would be disposed of in the Z-Area Saltstone Disposal Units.

2.1.6 CASTOR Cask Disposition

The empty CASTOR casks and the inner canisters in which the pebbles would be shipped from Germany would be managed as LLW; their disposition is not shown in the Figures 2-4 and 2-7, which depict the process flow for the pebbles and disposition of the principal waste streams. Disposition of the casks would depend on the alternative or option selected for disposition of the kernels. Under the H-Area Alternative, LEU Waste and LEU/Thorium Waste Options, the casks could be used for waste transportation and disposal. Under the Vitrification Option and the L-Area Alternative, the empty casks would be disposed as LLW.

2.1.6.1 Cask Disposition under the H-Area Alternative

Under the Vitrification Option, the primary waste streams generated by processing the spent nuclear fuel kernels would be processed through the SRS Liquid Nuclear Waste Facilities for disposal as HLW glass and LLW saltstone. Under this option, the inner canisters could be replaced inside the casks and disposed of as LLW at the SRS E-Area, offsite at NNSS, or offsite at a commercial LLW disposal facility. The casks, containing the canisters, would result in a disposal waste volume of approximately 67,000 cubic feet.

Alternatively, although DOE has not identified any future use for the casks, it is possible that they could be reused by another entity for the storage and/or transport of radioactive materials. As a result of their past use, the metal from which the casks are constructed may be activated or the casks may have some level of internal radioactive contamination. If there were interest in reusing the casks, DOE could transfer the casks to another entity (e.g., a company or government agency) that demonstrated the qualifications and necessary licenses and permits to assume ownership of the casks. If the casks were reused, the inner canisters would be disposed of separately as LLW in the SRS E-Area trenches. The canisters would represent approximately 8,000 cubic feet of LLW.

Under the LEU Waste and LEU/Thorium Waste Options, the casks could be reused for disposal of the grouted LLW form. The waste containers used in the cementation process would be sized to fit within the casks, and the casks and grouted waste forms would be disposed of together at the SRS E-Area, offsite at NNSS, or offsite at a commercial LLW disposal facility. The volume of the casks with LLW grout would be the same as for the casks with canisters, that is, approximately 67,000 cubic feet. Under these options, the inner canisters would be disposed separately in the SRS E-Area trenches, and would represent approximately 8,000 cubic feet of LLW.

2.1.6.2 Cask Disposition under the L-Area Alternative

The casks and the inner canisters would be managed as described in Section 2.1.6.1 under the H-Area Alternative, either disposed as described for the Vitrification Option or considered for reuse. If disposed, the casks and canisters would result in a disposal waste volume of approximately 67,000 cubic feet.

If the casks were reused, the inner canisters would be disposed of separately as LLW in the SRS E-Area trenches. The canisters would represent approximately 8,000 cubic feet of LLW.

2.2 NO ACTION ALTERNATIVE

Under the No Action Alternative, the spent nuclear fuel containing U.S.-origin uranium from the AVR and THTR would not be transported to the United States for management and disposition. The spent nuclear fuel would remain in storage in Germany and the impacts described in Chapter 4 of this Draft EA would not occur.

2.3 ALTERNATIVES CONSIDERED BUT DISMISSED FROM DETAILED ANALYSIS

Additional alternatives and technology options were identified prior to and during the development of this EA. For the reasons discussed in this section, these alternatives and options were not included in the detailed analysis.

2.3.1 Carbon Removal Technologies

The Savannah River National Laboratory undertook a feasibility study to systematically evaluate potential technologies for processing the spent nuclear fuel from Germany (SRNL 2014a). A number of technologies for separating the spent nuclear fuel kernels from the graphite (carbon) pebbles were considered and eliminated from detailed analysis for technical reasons. Processes considered but determined to be unacceptable include direct dissolution, oxidation in a fluidized bed, and mechanical removal. Direct dissolution of the carbon using nitric acid was previously evaluated. Cursory testing showed extremely wide variability of success due to differences in carbon fabrication and concluded the direct dissolution method was unreliable (ORNL 1973). Historical experience with oxidation in a fluidized bed shows that fission product volatilization presents emissions problems and ash residue presents disposition problems (SRNL 2014a). Mechanical removal technologies, such as crushing or grinding, create fines (small particles) that present a problem in the downstream processing steps.

2.3.2 Direct Spent Fuel Disposal Alternative

The concept for this alternative would be to dispose of the pebbles containing the uranium without processing. Conceptually, CASTOR casks containing the pebbles would be accepted for disposal as currently packaged. Alternatively, the pebbles would be removed from the CASTOR casks and inner canisters and repackaged in a different type of spent nuclear fuel shipping and disposal cask. This alternative was dismissed from further consideration because the CASTOR cask is not a qualified disposal container nor is there an existing spent nuclear fuel cask qualified for disposal of this type of spent fuel. Also, compared to the alternatives and options evaluated in this EA, this alternative results in a much larger quantity of material (about 20 times more than the 101 HLW canisters that would be generated under the Vitrification Option) requiring deep geologic disposal.

2.3.3 Down Blending to LEU for Use as Reactor Fuel Option

This option for disposition of the uranium from the spent fuel was originally included in the NOI (79 FR 32256). It is similar to the LEU Waste Option with respect to dissolution and processing through the H-Canyon solvent extraction process to recover uranium. This option would likely involve additional solvent extraction processing to increase the purity of the uranium. Following dissolution and purification, the resulting uranium would be down blended with LEU or depleted uranium to an enrichment appropriate for use in commercial nuclear power reactors. The LEU would be converted to an oxide and packaged for storage pending transfer to a fuel fabrication vendor. The spent nuclear fuel from Germany has a high uranium-232 content and a higher dose rate than other LEU due to the short-lived daughter products which emanate very energetic and penetrating gamma radiation, requiring extensive gamma shielding for fuel fabrication. This requirement makes this material unattractive for this use.

2.3.4 Uranium Solidification in the Uranium Stabilization Facility

The Uranium Stabilization Facility (USF), located in a radiologically clean area on the first level of H-Canyon was established as a facility for stabilizing uranium, a different process than the solidification being considered under the H-Canyon Alternative, LEU Waste and LEU/Thorium Waste Options. Based on other cementation plants, the layout of a solidification facility for uranium or uranium/thorium would require an enclosed area of about 40 feet by 75 feet for the cementation stations, drum movement, and loading. In addition, a similarly sized area would be required to house the dry material feed tanks and the ventilation and exhaust system. The layout of USF is significantly smaller than this so this area of H-Canyon was not considered a reasonable location for the solidification facility.

3 AFFECTED ENVIRONMENT

In accordance with the Council on Environmental Quality's National Environmental Policy Act (NEPA) regulations (40 *Code of Federal Regulations* [CFR] Parts 1500 through 1508) and DOE's NEPA implementing procedures (10 CFR Part 1021), areas that could be affected by the Proposed Action are succinctly described in this chapter. This chapter includes descriptions of: 1) the global commons that would be traversed by ships carrying the spent nuclear fuel; 2) Joint Base Charleston – Weapons Station, the seaport at which such ships would dock; and 3) SRS, the location in the United States at which the spent nuclear fuel would be stored and processed for disposition. The affected environment descriptions provide the context for understanding the environmental consequences described in Chapter 4 of this EA, and serve as baselines from which any potential environmental impacts can be evaluated. For this EA, each resource area that may be affected by the Proposed Action is described. The level of detail varies depending on the potential for impacts on each resource area.

3.1 GLOBAL COMMONS

The global commons includes the world's oceans that would be traversed by transport ships. The structural features of the oceans can be divided into the shore, continental shelf, continental slope and rise, basin (or abyssal plain), and mid-oceanic ridges. The shore region is that portion of the land mass that has been modified by oceanic processes. Providing some of the richest fisheries known, the continental shelf extends seaward from the shore and is characterized by a gentle slope of about 1:500. At the end of the shelf, the steepness of the slope first increases to about 1:20 (the continental slope), and then reduces (the continental rise). The ocean basin constitutes about 75 percent of the ocean bottom, ranging in depth from about 9,840 feet to 19,700 feet (3,000 meters to 6,000 meters). The deepest areas of the ocean basins are the deep sea trenches, contrasted by the mid-oceanic ridges, which provide relatively high points on the ocean bottom (DOE 1996a).

Seawater within the oceans is a complex solution of minerals, salts, and elements. Naturally occurring radionuclides are present in seawater and marine organisms at concentrations greater than in terrestrial ecosystems (DOE 1996a). The inventory of natural radionuclides in the oceans is about 5×10^{11} curies. Radionuclides have also been released into the oceans from nuclear weapons testing, radioactive waste disposal, and accidents. It is estimated that the total input of radionuclides from human activities represents somewhat less than 1 percent of the natural radioactive material present in the oceans (DOE 2006c).

Biologically, the characteristics of ocean organisms dramatically change with depth, largely a result of the decrease in the amount of light and changes in the wavelength of light penetrating to a given depth. Deep-sea bottom dwellers, or benthos, are highly diverse, with many taxonomic groups being represented by more species than most shallow-water communities. Yet the number of individual organisms in a given area decreases in the deep seas and this, together with a tendency for the average size of the organisms to also decrease, results in a dramatic reduction in biomass on the deep ocean floor (DOE 2009a).

The United States has jurisdiction over 122 endangered and threatened marine species, including 32 foreign species[20] (NOAA 2014a). The Atlantic Ocean, which would be traversed by the ships transporting the spent nuclear fuel under the proposed action, contains some of the world's most productive fisheries, located on the continental shelves and marine ridges, and contributes 13 percent of world-wide aquaculture and commercial catches. Major commercial fish species include menhaden, herring, cod, mackerel, and pollock (NOAA 2013a). Marine species that live in the Atlantic Ocean and are on the Federal endangered species list include whale species [e.g., North Atlantic right whale (*Eubalaena glacialis*), sperm whale (*Physeter macrocephalus*)], all six species of sea turtles [loggerhead (*Caretta caretta*), leatherback (*Dermochelys coriacea*), green (*Chelonia mydas*), hawksbill (*Eretmochelys imbricate*), Kemp's ridley (*Lepidochelys kempii*), and olive ridley (*Lepidochelys olivacea*)], as well as the West Indian manatee (*Trichechus manatus*) (NOAA 2014a). They are found in both the northern and southern parts of the Atlantic Ocean and most of these marine species have the potential to occur around Joint Base Charleston – Weapons Station, the U.S. seaport evaluated in this EA.

Effective August 11, 2014, the National Marine Fisheries Service (NMFS) and the U.S. Fish and Wildlife Service designated critical habitat[21] for the loggerhead sea turtle within the Northwest Atlantic Ocean Distinct Population Segment and nesting beaches off the coast of North Carolina, South Carolina (including Charleston beaches), Georgia, Florida, Alabama, and Mississippi (79 FR 39855, 79 FR 39755). Mating season occurs in late March to early June followed by nesting season between late April and early September. After about a two-month incubation period, hatching occurs between late June and mid-November. The greatest threat to the loggerhead sea turtle is incidental capture (NOAA 2014b).

The North Atlantic right whale is also protected internationally under the Convention for the Regulation of Whaling and is designated a "depleted" species under the Marine Mammal Protection Act. There are currently about 450 right whales in the North Atlantic, with ship strikes and entanglement in fishing gear being the most common human cause of severe injury or death (NOAA 2013b). NMFS designated critical habitat for the North Atlantic right whale identified in areas off the coast of Massachusetts and off the coasts of Georgia and Florida (59 FR 28805).

The Maritime Safety Committee of the International Maritime Organization (IMO) adopted a mandatory ship reporting system that became effective in 1999. This system requires ships to report whale sightings in the major shipping lanes off the southeastern coast of the United States from November 15 to April 15 so as to include the calving season for the right whales in this area. The system operates throughout the year on the northeastern coast, where the whales have been sighted year-round (IMO 1998). Consistent with the IMO requirement, before entering an area routinely inhabited by right whales, the U.S. Coast Guard requires ships exceeding 270 gross metric tons (300 tons) to contact their Mandatory Ship Reporting System to report the ship's name,

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[20] Foreign species refers to species that occur exclusively in foreign waters. Under the Endangered Species Act, all endangered and threatened species are listed, regardless of where they are found.

[21] Critical habitat is identified as habitat essential to the conservation of an endangered or threatened species. Listed species and their habitat are protected under the Endangered Species Act, which forbids all actions that result in illegal "take" [16 U.S.C. 1531(19)], including injury through habitat alteration or destruction. The Act also prohibits Federal actions that may result in adverse modification of habitat [16 U.S.C. 1536(a)].

call sign, location, course, speed, destination, and route. This system reduces the likelihood of a ship striking a right whale by providing ships in the area with data on the most recent whale sightings and whale avoidance procedures (DOE 2006c). To further reduce the likelihood of ships colliding with right whales, on October 10, 2008, NMFS established regulations implementing speed restrictions for vessels having lengths equaling or exceeding 65 feet (19.8 meters) (73 FR 60173). These regulations apply within designated areas off the East Coast of the United States at certain times of the year; for the areas off the coasts of Florida and South Carolina, the restrictions apply from certain dates in November through certain dates in April (50 CFR 224.105).[22,23]

3.2 U.S. PORT OF ENTRY, JOINT BASE CHARLESTON - WEAPONS STATION

In October 2010, Charleston Naval Weapons Station and Charleston Air Force Base were combined to become Joint Base Charleston, as recommended by the 2005 Base Closure and Realignment Commission (Military OneSource 2014). For purposes of this EA, DOE is evaluating Joint Base Charleston – Weapons Station, which would be the port of entry to the United States for the spent nuclear fuel. Joint Base Charleston – Weapons Station is approximately 10 miles (16 kilometers) north of metropolitan Charleston. The principal shipping terminals at Joint Base Charleston - Weapons Station are located along the west bank of the Cooper River, north of the city of North Charleston and about 19 miles (31 kilometers) upriver from the Atlantic Ocean. Charleston is the largest port city in South Carolina, and the greater Charleston area is a major seaport on the east coast of the United States. The Charleston area highway system includes Interstates 26 and 526 and U.S. Routes 17 and 52 (DOE 2009a). The region around Charleston and Joint Base Charleston - Weapons Station is shown on **Figure 3-1**.

Joint Base Charleston – Weapons Station encompasses over 17,000 acres (6,900 hectares) of land with 10,000 acres (4,000 hectares) of forest and wetlands, 16 miles (26 kilometers) of waterfront, four deep-water piers (including piers capable of unloading transport containers directly from ships), 38 miles (61 kilometers) of railroad and 292 miles (470 kilometers) of road. The base provides ordnance storage capability and other material supply and support functions and has the ability to load and unload cargo directly between vehicles and ships (MARCOA Publishing, Inc. 2015).

[22] Regulations restricting ship speed in designated areas off the East Coast do not apply to "U.S. vessels owned or operated by, or under contract to, the Federal Government."

[23] The section of the Code of Federal Regulations limiting vessel speed in designated areas off the East Coast during certain times of the year had a sunset clause of December 9, 2013. A final rule promulgated by the National Marine Fisheries Service removed the sunset clause such that the speed restrictions remain in force (78 FR 73726).

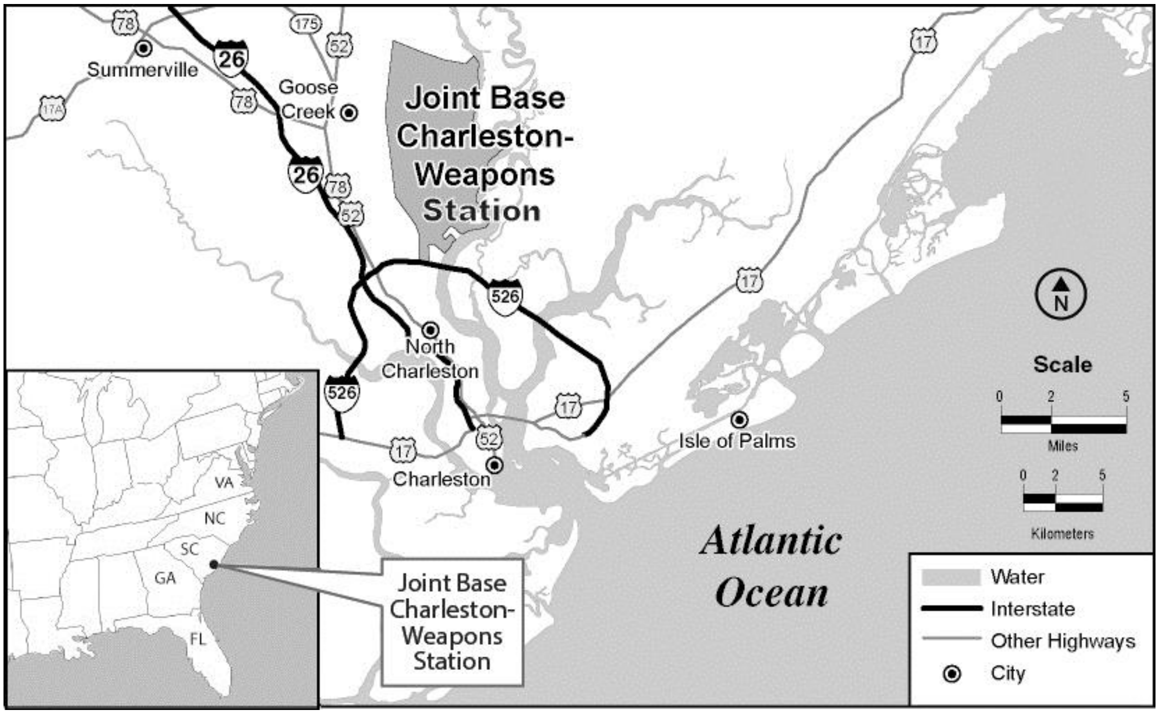


Figure 3-1: Region Around Joint Base Charleston - Weapons Station

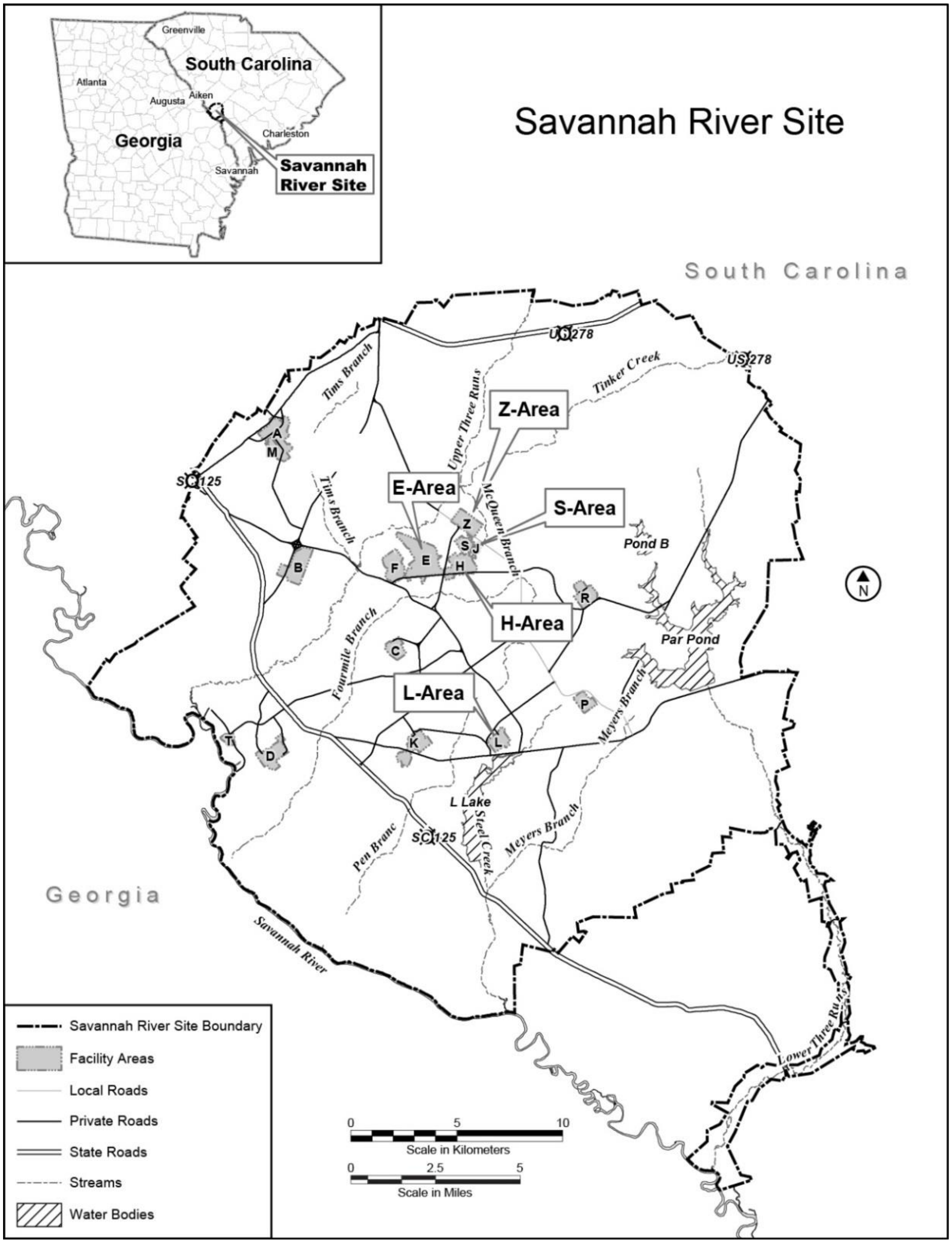
According to the 2010 census, approximately 773,000 people lived within 50 miles (80 kilometers) of the docks at Joint Base Charleston – Weapons Station; approximately 737,000 people lived within 50 miles (80 kilometers) of the Charleston harbor through which vessels pass to enter the Cooper River. Joint Base Charleston – Weapons Station supports approximately 17,400 military, civilian and contract employees in addition to providing student housing to more than 2,800 enlisted nuclear power students, and privatized on-base housing to about 800 military families (MARCOA Publishing, Inc. 2015). The population in the area is growing. The natural background radiation dose to an average individual in the population near Joint Base Charleston – Weapons Station was assumed to be the same as that to an average individual in the United States, i.e., approximately 311 millirem per year (NCRP 2009).

Joint Base Charleston - Weapons Station offers a secure site conducive to transferring spent nuclear fuel from ships to transport vehicles, including rail cars. Joint Base Charleston – Weapons Station routinely receives marine shipments of spent nuclear fuel under the U.S. Foreign Research Reactor Spent Nuclear Fuel Acceptance Program. Since this program was established in 1996, over 60 shipments of spent nuclear fuel have been received in the United States, most of which were received at Joint Base Charleston – Weapons Station (NNSA 2013). The spent fuel casks have been offloaded from ships to trucks or rail cars, and transported to DOE facilities (DOE 2009a).

3.3 SAVANNAH RIVER SITE

Located in southwestern South Carolina, SRS occupies an area of 198,344 acres (80,268 hectares) in a generally rural area about 25 miles (40 kilometers) southeast of Augusta, Georgia, and 12 miles (19 kilometers) south of Aiken, South Carolina, the nearest population centers. It is bordered by the Savannah River to the southwest and includes portions of three South Carolina counties:

Aiken, Allendale, and Barnwell. **Figure 3-2** is a map of SRS. SRS is a controlled area, public access being limited to through traffic on State Highway 125 (SRS Road A), U.S. Highway 278 (SRS Road 1), and the CSX railway line (DOE 2015a, SRNS 2013).



Source: DOE 2015

Figure 3-2: **Savannah River Site Map**

The proposed alternatives evaluated in this EA would be primarily conducted within H- and L-Areas. As shown in Figure 3-2, H-Area covers 395 acres (160 hectares) and is located near the center of SRS, 6.8 miles (11 kilometers) from the site boundary (DOE 2002, 2012a). H-Area contains nuclear, chemical, industrial, administrative, laboratory, and storage facilities, and includes H-Canyon, HB-Line, and the H-Area Tank Farm (DOE 2006a). H-Canyon was constructed in the early 1950s and began operations in 1955. The Enriched Uranium Disposition Mission, a non-proliferation program that renders HEU no longer useable for nuclear weapons, is the primary mission of the H-Canyon Complex. H-Canyon and its ancillary facilities are used to dissolve, purify and down blend surplus highly enriched uranium and aluminum-clad foreign and domestic research reactor fuel to produce a low-enriched uranium solution suitable for conversion to commercial nuclear reactor fuel. A secondary mission for H-Canyon is surplus plutonium disposition (DOE 2012a).

L-Area is located in the south-central part of SRS, approximately 5.7 miles (9.2 km) from the site boundary. L-Area was initially constructed as a nuclear reactor for use as a nuclear material production facility in the 1950s. The reactor was shut down in 1968, and then restarted for a short period in the 1980s. In the 1990s, the function of the facility was changed to storing nuclear material. In addition to the L-Reactor Facility, now called the Material Storage Facility, the facility contains areas still referred to as the Assembly Area and the Purification Area. There are also a Personnel Wing, the Moderator Storage Area, and waste staging areas. The current mission for the L-Area Facility is to provide for the safe receipt, storage, handling, and shipping of spent nuclear fuel and other special nuclear materials. Spent fuel assemblies are received from research reactors in the United States, other DOE facilities, and from foreign research reactors. The Material Storage Facility also receives, stores, handles, and ships moderator and fissile/fissionable material from various DOE facilities (SRNS 2014a).

The main waste management facilities that would be used to support the proposed activities are located in E-, S-, and Z-Areas. E-Area is located near the center of SRS to the west of H-Area. E-Area comprises approximately 330 acres (134 hectares) and includes the Old Burial Ground, Mixed Waste Management Facility, transuranic waste pads, and E-Area Vaults. E-Area receives solid LLW, TRU waste, and mixed waste from across SRS (DOE 2015a).

S-Area is located near the center of SRS between the H- and Z-Areas. This area is approximately 272 acres (110 hectares) in size (DOE 2015a). S-Area facilities are used to process radioactive liquid waste for geologic disposal. Facilities include the Defense Waste Processing Facility (DWPF), Salt Waste Processing Facility (currently under construction), glass waste storage facilities, and typical support structures such as administrative office buildings, maintenance and repair shops, and warehouses to store equipment and material. DWPF accepts waste from the H-Tank Farm and will accept waste from the Salt Waste Processing Facility, when the latter facility becomes operational. The Salt Waste Processing Facility will be used to separate the high-activity liquid waste portion of the H-Tank Farm salt solution from the low-activity liquid waste. High-activity liquid waste will be sent to DWPF for incorporation into glass in stainless steel canisters, and safe storage in S-Area pending disposition at a geologic repository. Low-activity liquid waste from the Salt Waste Processing Facility will be sent to the Saltstone facilities for disposal (DOE 2006a).

Located near the center of SRS, Z-Area is approximately 180 acres (72.8 hectares) in size (DOE 2001a). The Saltstone facilities, comprising the Saltstone Production Facility and the Saltstone Disposal Facility, are located in Z-Area (DOE 2006b).

3.3.1 Meteorology, Air Quality, and Noise

3.3.1.1 Meteorology

SRS has a temperate climate with short, mild winters and long, humid summers. The climate is frequently affected by warm, moist maritime air masses. Recent data are presented in the *Savannah River Site Annual Meteorology Report for 2012* (SRNL 2013). The historical average temperature is 64.6 degrees Fahrenheit (°F) (18.1 degrees Celsius [°C]) and the historical average annual precipitation is 41.2 inches (104 centimeters) (SRNL 2013). Temperatures vary from an average daily minimum of 39.2°F (4.0°C) in January to an average daily maximum of 91.7 °F (33.2 °C) in July (SRNL 2013).

Precipitation is distributed fairly evenly throughout the year, with the highest in summer and the lowest in autumn. The average annual windspeed at SRS is 3.9 miles per hour (1.7 meters per second) (SRNL 2013). The maximum windspeed at SRS (highest 15-minute average) is 19.9 miles per hour (8.9 meters per second) (SRNL 2013). Annual wind rose plots for the Central Climatology Tower at SRS for 2012 are provided in **Figure 3-3**. Typical wind direction patterns for the 200 foot (61-meter) elevation consist of higher frequencies of wind from the northeast section and the south to west sections. Typical variation of winds with elevation show higher frequencies of east to southeast winds and lower frequencies of south to southwest winds nearer the ground (SRNL 2013).

Damaging hailstorms and flooding rarely occur in Aiken County (NCDC 2014). The average annual snowfall is 2 inches (5 centimeters) (Aiken County Government 2014). Twenty-one tornadoes were reported in Aiken County between January 1950 and August 2014. There are typically several occurrences of high winds every year, mostly associated with thunderstorms (NCDC 2014). Hurricanes struck South Carolina 37 times during the period from 1700 to 2014, which equates to an average recurrence frequency of one hurricane every 8.5 years. A hurricane-force wind of 75 miles per hour (34 meters per second) has been observed at SRS only once, during Hurricane Gracie in 1959 (DOE 2002).

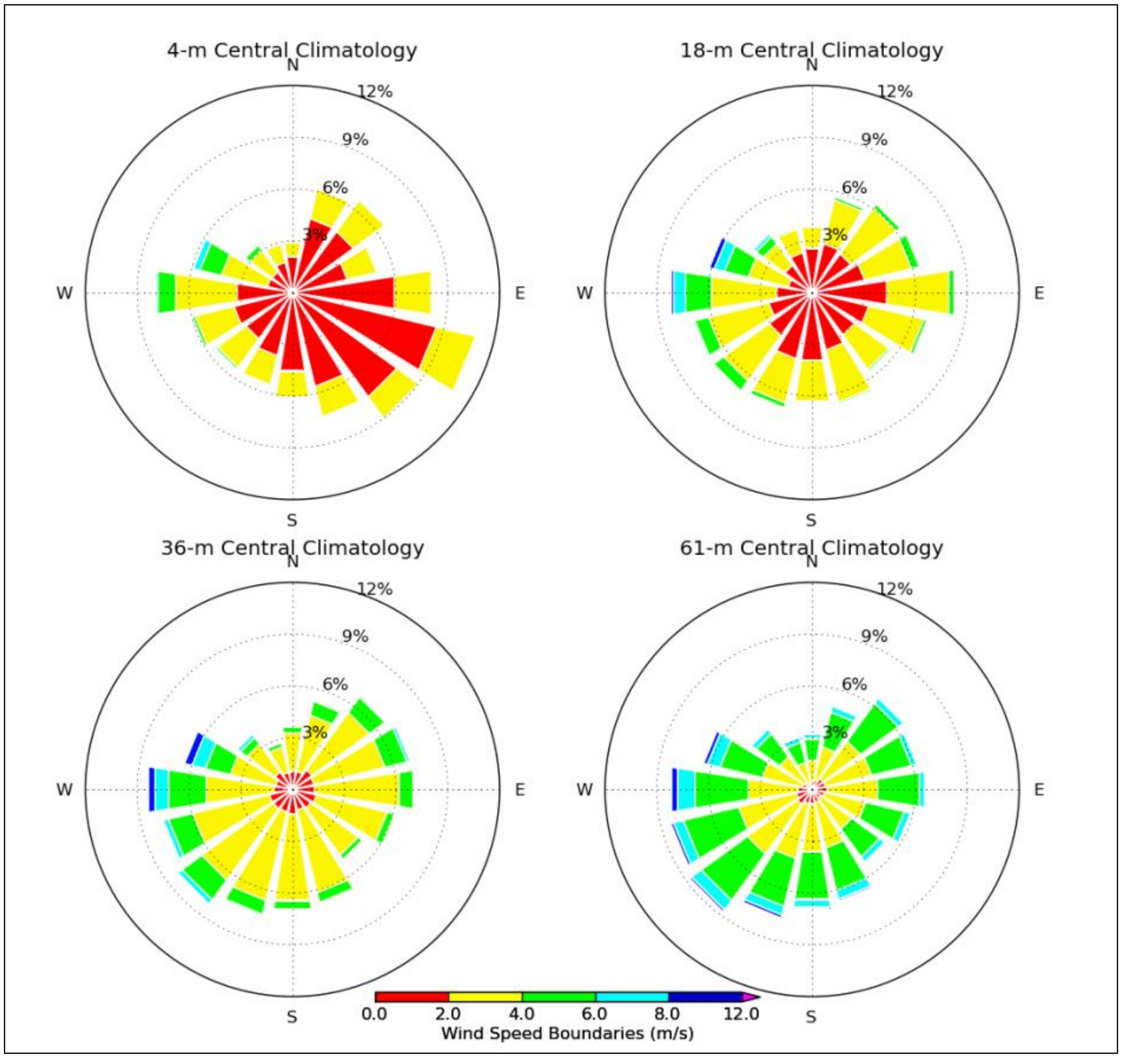


Figure 3-3: Annual Wind Rose Plots for 2012, Central Climatology Tower, All Levels

Notes: Wind rose plot depicts the frequency of occurrence of wind direction sector (direction from which the wind blows) by speed category.

Source: SRNL 2013

3.3.1.2 Air Quality

Air pollutants are any substances in the air that could harm humans, animals, vegetation, or structures, or that could unreasonably interfere with the comfortable enjoyment of life and property. Air quality is affected by air pollutant emission characteristics, meteorology, and topography.

SRS is located near the center of the Augusta-Aiken Interstate Air Quality Control Region (AQCR) 53, which includes 6 counties in South Carolina and 13 in Georgia. **Table 3-1** provides baseline annual emissions data obtained from EPA's 2011 National Emissions Inventory (NEI) for Aiken County and the Augusta –Aiken AQCR (EPA 2014a). These data are the latest available. The data include emissions from point sources, area sources, and mobile sources. *Point sources* are stationary sources that can be identified by name and location. *Area sources* are stationary sources from which emissions are too low to track individually, such as a home or small office building; or a diffuse stationary source, such as wildfires or agricultural tilling. *Mobile sources* are any kind of vehicle or equipment with gasoline or diesel engine, an airplane, or a ship. None of the areas within SRS or its surrounding counties are designated as nonattainment areas with respect to the National Ambient Air Quality Standards for criteria air pollutants (EPA 2014a).

Table 3-1: Baseline Criteria Pollutant Emissions Inventory for Aiken County and AQCR 53

Region	Criteria Pollutant Emissions (tons/year)					
	CO	NO_2	PM_{10}	$PM_{2.5}$	SO_2	VOC
Aiken Co.	49,790	7,646	23,730	7,217	5,020	40,128
AQCR 53	263,720	47,378	150,427	40,231	16,404	369,822

CO = carbon monoxide; NO_2 = nitrogen dioxide; PM_{10} and $PM_{2.5}$ = particulate matter with a diameter of less than or equal to 10 microns and 2.5 microns, respectively; SO_2 = sulfur dioxide; VOC = volatile organic compounds

Source: EPA, 2014a

There are no Prevention of Significant Deterioration Class I areas within 62 miles (100 kilometers) of SRS (NPS 2014). Class I areas are areas in which very little increase in air pollution is allowed due to the pristine nature of the area.

The primary sources of air pollutants at SRS are the biomass boilers in K- and L-Areas, diesel-powered equipment throughout SRS, the Defense Waste Processing Facility (DWPF), soil vapor extractors, groundwater air strippers, the Biomass Cogeneration Facility and back-up oil-fired boiler on Burma Road, and various other processing facilities. Other emissions and sources include fugitive particulates from vehicles and controlled burning of forested areas, as well as temporary emissions from various construction-related activities (DOE 1999; NRC 2005; SRNS 2011, DOE 2015a).

Four biomass boilers and one new oil-fired auxiliary boiler are operated under a 40 CFR Part 70 Prevention of Significant Deterioration permit process because of carbon monoxide emissions (SRNS 2013). The current SRS Title V Part 70 operating permit expired in June 2012. SRS submitted a renewal application but has not yet received the new permit from the South Carolina Department of Health and Environmental Control (SCDHEC). Until the permit is renewed, SRS continues to operate in accordance with requirements of the current permit (SRNS 2013).

SRS is required by the Title V Part 70 Operating Permit to demonstrate compliance through air dispersion modeling and submittal of an annual emissions inventory of air pollutant emissions. **Table 3-2** shows the total air pollutant emission estimates for all SRS permitted sources as determined by the air emissions inventory conducted for the last five years. SCDHEC review of the emissions has found that SRS sources operated in compliance with permitted emission rates and the ambient air quality standards (SRNS 2014b).

Table 3-2: Savannah River Site Estimated Nonradiological Air Pollutant Emissions, 2009-2013

Pollutant	Emissions (tons/year)					
	2009	2010	2011	2012	2013	Average
SO_2	4,000	4,110	4,560	953	6.8	2,726
PM_{10}	264	637	142	18	9.1	214
$PM_{2.5}$	222	136	427	16	7.2	162
CO	40.7	44.6	125	52	21.7	56.8
VOC	65	45	46	40	41.5	47.5
NO_2	1,790	2,060	2,060	621	268	1360
Pb	0.034	0.0391	0.0166	0.00064	0.0047	0.0190
HF	12.2	12.2	12.3	2	0.0025	7.74

CO = carbon monoxide; NO_2 = nitrogen dioxide; PM_{10} and $PM_{2.5}$ = particulate matter with a diameter of less than or equal to 10 microns and 2.5 microns, respectively; SO_2 = sulfur dioxide; VOC = volatile organic compounds, Pb = lead and lead compounds, HF = hydrogen fluoride

Source: SRNS 2014b

Table 3-3 presents the applicable ambient air quality standards and ambient air pollutant concentrations for sources at SRS. These concentrations are based on potential emissions (SRNS 2011). Concentrations shown in Table 3-3 attributable to SRS are in compliance with applicable guidelines and regulations.

Recent data from nearby ambient air monitors in Aiken and Richland Counties in South Carolina are presented in **Table 3-4**. The data indicate that the National Ambient Air Quality Standards for particulate matter, lead, ozone, sulfur dioxide, and nitrogen dioxide are not exceeded in the area around SRS (EPA 2014a).

Table 3-3: Comparison of Potential Ambient Air Pollutant Concentrations from Existing Savannah River Site Sources with Applicable Standards or Guidelines

Pollutant	Averaging Period	More Stringent Standard or Guideline (micrograms per cubic meter)[a]	Estimated concentration (micrograms per cubic meter)
Criteria Pollutants			
Carbon monoxide (CO)	8 hours	10,000[b]	292
	1 hour	40,000[b]	1,118.2
Nitrogen dioxide (NO_2)	1-hour	188[c]	
	Annual	100[b]	42.1
Ozone	8 hours	147[c]	(d)
PM_{10} [e]	24 hours	150[b]	50.7
$PM_{2.5}$	Annual	12[c]	(f)
	24 hours (98th percentile over 3 years)	35[b]	(f)
Sulfur dioxide (SO_2)	1 hour	197[c]	155.1
	3 hours	1,300[b]	723
Lead (Pb)	Rolling 3-month average	0.15[b]	0.11
Other Regulated Pollutants			
Gaseous fluoride (HF)	30 days	0.8[g]	0.03
	7 days	1.6[g]	0.21
	24 hours	2.9[g]	0.23
	12 hours	3.7[g]	0.35
Hazardous and Other Toxic Compounds			
Benzene	24 hours	150[g]	0.082

PM_n = particulate matter less than or equal to *n* microns in aerodynamic diameter.

[a] The more stringent of the Federal and state standards is presented if both exist for the averaging period. Methods of determining whether standards are attained depend on pollutant and averaging time. National Ambient Air Quality Standards (EPA 2012), other than those for ozone, particulate matter, and lead, and those based on annual averages, are not to be exceeded more than once per year. The 8-hour ozone standard is attained when the 3-year average of the annual fourth-highest daily maximum 8-hour average concentration is less than or equal to the standard. The 24-hour PM_{10} standard is attained when the expected number of days with a 24-hour average concentration above the standard is less than or equal to 1. The 24-hour $PM_{2.5}$ standard is attained when the 3-year average of the 98th percentile 24-hour averages is less than or equal to the standard. The annual $PM_{2.5}$ standard is attained when the 3-year average of the annual means is less than or equal to the standard.

[b] Federal and state standard.

[c] Federal standard.

[d] No concentration reported.

[e] EPA revoked the annual PM_{10} standard in 2006.

[f] $PM_{2.5}$ values are not yet available from the modeling for the Title V permit application because the modeling methodology for $PM_{2.5}$ is still under discussion with SCDHEC. Currently, the SCDHEC policy is to use demonstration of PM_{10} compliance as a surrogate for $PM_{2.5}$ compliance (SRNS 2011).

[g] State standard.

Source: EPA 2012; SCDHEC 2012; SRNS 2011.

Table 3-4: Ambient Air Quality Standards and Monitored Levels in the Savannah River Site Vicinity

Pollutant	Averaging Time	Ambient Standard (micrograms per cubic meter)	Concentration (micrograms per cubic meter)	Location
Carbon monoxide (CO)	8 hours	10,000	788[a]	Richland County, South Carolina
	1 hour	40,000	1,240[a]	Richland County, South Carolina
Nitrogen dioxide (NO_2)	1 hour	188	34	Richland County, South Carolina
	Annual	100	6.6[a]	Aiken County, South Carolina
Ozone	8 hours	147	125[a]	Aiken, South Carolina
PM_{10}	24 hours	150	35[a]	Richland County, South Carolina
$PM_{2.5}$	Annual	12	9.4[a]	Aiken, South Carolina
	24 hours (98th percentile over 3 years)	35	21.1[b]	Aiken, South Carolina
Sulfur dioxide (SO_2)	1 hour	197	178.6[a]	Richland County, South Carolina
	3 hours	1,300	39.3[a]	Barnwell, South Carolina
Lead (Pb)	Calendar quarter	0.15	0.005[c]	Richland County, South Carolina

PM_n = particulate matter less than or equal to *n* microns in aerodynamic diameter.

[a] 2013 data.

[b] 2013 3-year average.

[C] 2013 data. Rolling 3 month average not available. Annual 1st-4th maxes provided.

Note: EPA recently promulgated 1-hour standards for nitrogen dioxide and sulfur dioxide and a rolling 3-month average standard for lead for which monitoring data are not yet available. The nearby monitor in Barnwell County has been discontinued.

Source: EPA 2014a

The "natural greenhouse effect" is the process by which part of the terrestrial radiation is absorbed by gases in the atmosphere, thereby warming the Earth's surface and atmosphere. This greenhouse effect and the Earth's radiative balance are affected largely by water vapor, carbon dioxide, and trace gases, all of which are absorbers of infrared radiation and commonly referred to as "greenhouse gases." Other trace gases include nitrous oxide, chlorofluorocarbons, methane, and sulfur hexafluoride. EPA reporting currently only includes carbon dioxide, methane, and nitrous oxide. Annual greenhouse gas (GHG) emissions data for both Aiken County and the Augusta-Aiken AQCR from the EPA's 2011 NEI are provided in **Table 3-5**.

Table 3-5: Baseline Greenhouse Gas Emissions Inventory for Aiken County and AQCR 53

Region of Interest	Greenhouse Gas (tons/year)			
	CO_2	CH_4	N_2O	CO_2e
Aiken Co.	1,265,157	236	36	1,282,823
AQCR 53	6,760,853	30,040	22,777	6,869,310

CO_2 = carbon dioxide; CH_4 = methane; N_2O = nitrous oxide; CO_2e = carbon dioxide equivalent
Source: EPA 2014b

Based on the number of employee vehicle trips estimated from employment at SRS and fuel and electricity use, emissions of carbon dioxide attributable to SRS activities were estimated to be 0.502 million metric tons per year (Messick 2012). Carbon dioxide emissions associated with ongoing transportation of materials and goods at SRS are expected to be substantially less than that for SRS employee vehicle trips and fuel and electricity use and no further analysis is warranted.

SRS has made strides toward reducing GHG emissions. According to the *SRS Environmental Report for 2013* (SRNS 2014b), SRS has greatly reduced GHG emissions by transferring to a biomass-based energy supply versus the previous coal-based supply. GHG reduction of 75.2 percent was realized in FY 2013 due to the operation of the existing biomass plants in K- and L-Areas and the recent addition of the Ameresco Biomass Co-generation Facility near F-Area.

3.3.1.3 Noise

Noise is any unwanted sound that interferes or interacts negatively with the human or natural environment. Noise may disrupt normal activities, diminish the quality of the environment, or if loud enough, cause discomfort and even hearing loss.

Major noise sources at SRS occur primarily in developed or active areas and include various industrial facilities, equipment, and machines (e.g., cooling systems, transformers, engines, pumps, boilers, steam vents, public address systems, construction and materials-handling equipment, and vehicles). Major noise emission sources outside of these active areas consist primarily of vehicles and rail operations. Existing SRS-related noise sources of importance to the public are those related to transportation of people and materials to and from the site, including trucks, private vehicles, helicopters, and trains (DOE 2015a). Another important contributor to noise levels is traffic to and from SRS along access highways through the nearby towns of New Ellenton, Jackson, and Aiken, South Carolina.

Most industrial facilities at SRS are far enough from the site boundary that noise levels at the boundary from these sources would not be measurable or would be barely distinguishable from background levels.

3.3.2 Human Health

Public and occupational health and safety issues at SRS are potentially adverse effects on human health that result from acute and chronic exposure to ionizing radiation and hazardous chemicals.

3.3.2.1 Radiological Exposure and Risk

General Site Description

Major sources and levels of background radiation exposure to individuals in the vicinity of SRS are assumed to be the same as those to an average individual in the U.S. population. These are shown in **Table 3-6**. Background radiation doses are unrelated to SRS operations and are expected to remain constant over time.

Table 3-6: Radiation Exposure of Individuals in the Savannah River Site Vicinity Unrelated to Savannah River Site Operations

Source	Effective Dose (millirem per year)
Natural background radiation[a]	
Cosmic and external terrestrial radiation	54
Internal terrestrial radiation	29
Radon-220 and -222 in homes (inhaled)	228
Other background radiation	
Diagnostic x-rays and nuclear medicine	300
Occupational	0.5
Industrial, security, medical, educational, and research	0.3
Consumer products	13
Total (rounded)	**620**

[a] An average for the United States.

Source: NCRP 2009

Releases of radionuclides to the environment from SRS operations provide another source of radiation exposure to individuals in the vicinity of SRS. The annual doses to the public from recent releases of radioactive materials (2009 through 2013) and the average annual doses over this 5-year period are presented in **Table 3-7**. These doses fall within limits established per DOE Order 458.1, *Radiation Protection of the Public and the Environment* (DOE 2011a) and are much lower than background radiation.

As shown, the average radiation dose received by a maximally exposed member of the public due to radiological releases from SRS operations from 2009 through 2013 is about 0.12 millirem. Using a risk estimator of 600 latent cancer fatalities (LCFs) per 1 million rem or person-rem (or 0.0006 LCFs per rem or person-rem) (DOE 2003), the annual average LCF risk to this receptor would be 7×10^{-8}. That is, the estimated probability of this person developing a fatal cancer at some point in the future from radiation exposure associated with 1 year of SRS operations is 1 in 14 million. (Note: It takes a number of years from the time of radiation exposure until a cancer manifests, if at all.)

Table 3-7: Annual Radiation Doses to the Public from Savannah River Site Operations for 2009–2013

Members of the Public	Year	Atmospheric Releases[a]	Liquid Releases[b]	Total [c, d]
Maximally exposed individual (millirem)[e]	2009	0.04	0.08	0.12
	2010	0.05	0.06	0.11
	2011[f]	0.06	0.08	0.14
	2012[f]	0.04	0.10	0.14
	2013[f]	0.04	0.05	0.09
	2009–2013 Average	0.05	0.07	0.12
Population within 50 miles (person-rem)[g]	2009	2.0	2.2	4.2
	2010	1.9	1.9	3.8
	2011[f]	1.2	1.8	3.0
	2012[f]	0.76	1.9	2.7
	2013[f]	2.2	1.2	3.4
	2009–2013 Average	1.6	1.8	3.4
Average individual within 50 miles (millirem)[h]	2009	0.0028	0.0025	0.0053
	2010	0.0024	0.0020	0.0044
	2011	0.0015	0.0019	0.0034
	2012	0.0010	0.0020	0.0030
	2013	0.0028	0.0013	0.0041
	2009–2013 Average	0.0021	0.0019	0.0041

[a] Maximally exposed individual doses from atmospheric releases are those reported for compliance with Clean Air Act regulations. DOE Order 458.1 (DOE 2011a) and Clean Air Act regulations in 40 CFR Part 61, Subpart H, establish a compliance limit of 10 millirem per year to a maximally exposed individual.

[b] Includes all water pathways, not just the drinking water pathway. Though not directly applicable to radionuclide concentrations in surface water or groundwater, an effective dose equivalent limit of 4 millirem per year for the drinking water pathway only is frequently used as a measure of performance. It is inspired by the National Primary Drinking Water Regulations maximum contaminant level for beta and photon activity that would result in a dose equivalent of 4 millirem per year (40 CFR 141.166).

[c] Total effective dose

[d] DOE Order 458.1 establishes an all-pathways dose limit of 100 millirem per year to individual members of the public.

[e] Beginning in the Savannah River Site Environmental Report for 2013 (SRNS 2014b), DOE uses a "representative person" as the receptor for analysis of impacts on an individual. The representative person receives a dose that is "representative of the more highly exposed individuals in the population." In this table, a distinction is not made between the MEI and representative person.

[f] Beginning with the *Savannah River Site Environmental Report for 2011* (SRNS 2012b), DOE includes the potential dose from use of Savannah River water for irrigation as part of the liquid pathway dose (not included in the doses in this table). Including the contribution from the irrigation pathway increases the average annual MEI dose by 0.09 millirem to 0.21 millirem, the offsite population dose by 2 person-rem to 5.4 person-rem.

[g] About 713,500 for 2009, based on 2000 census data, and about 781,060 for 2010–2013, based on 2010 census data. For liquid releases occurring from 2009 through 2013, an additional 161,300 water users in Port Wentworth, Georgia, and Beaufort, South Carolina (about 98 river miles downstream), are included in the assessment.

[h] Obtained by dividing the population dose by the number of people living within 50 miles of SRS for atmospheric releases; for liquid releases, the number of people includes water users who live more than 50 miles downstream of SRS.

Note: Sums and quotients presented in the table may differ from those calculated from table entries due to rounding. To convert miles to kilometers, multiply by 1.609.

Source: DOE 2010a; 2011b; 2012b; 2013a, 2014b.

Using the same risk estimator, annual emissions from normal operations during 2009-2013 are not expected to result in any excess latent cancer fatalities (LCFs) (calculated value of 0.001) in the population living within 50 miles (80 kilometers) of SRS. To put this number in perspective, it may be compared to the number of fatal cancers expected in the same population from all causes. The average annual mortality rate associated with cancer for the entire U.S. population is 186 per 100,000 people (CDC 2013). Based on this national mortality rate, the number of fatal cancers that would be expected to occur in the population living within 50 miles (80 kilometers) of SRS is 1,453 per year.

SRS workers receive the same dose as the general public from background radiation, but could also receive an additional dose from working in facilities with nuclear materials. **Table 3-8** presents the annual average individual and collective worker doses from SRS operations from 2009 through 2013. These doses fall within the regulatory limits of the DOE regulation, "Occupational Radiation Protection" (10 CFR Part 835). Using the risk estimator of 600 LCFs per 1 million person-rem (DOE 2003), the calculated average annual LCF risk of 0.08 in the workforce indicates a low probability of a single cancer fatality in the worker population.

Table 3-8: Radiation Doses to Savannah River Site Workers from Operations During 2009–2013

Occupational Personnel	From Onsite Releases and Direct Radiation by Year[a]					
	2009	2010	2011	2012	2013	Average
Average radiation worker (millirem)[b]	50	69	60	71	60	62
Total worker dose (person-rem)	109	180	150	145	89	134
Number of workers receiving a measurable dose	2,183	2,587	2,512	2,044	1471	2,159

a total effective dose equivalent

b No standard is specified for an "average radiation worker;" however, the maximum dose to an individual worker is 5,000 millirem per year (10 CFR Part 835). DOE's goal is to maintain radiological exposure as low as reasonably achievable (ALARA) and has therefore established an Administrative Control Level of 2,000 millirem per year. The SRS ALARA goal is to limit annual exposures to 500 millirem (DOE 2009b; SRS 2014).

Source: DOE 2010a; 2011b; 2012b; 2013a; 2014b.

3.3.2.2 Chemical Environment

The background chemical environment important to human health consists of the atmosphere, which may contain hazardous chemicals that can be inhaled; drinking water, which may contain hazardous chemicals that can be ingested; and other environmental media through which people may come in contact with hazardous chemicals (e.g., surface water during swimming, or food through ingestion). Hazardous chemicals can cause cancer and noncancerous health effects.

Effective administrative and design controls that decrease hazardous chemical releases to the environment and help achieve compliance with permit requirements (e.g., from the National Emission Standards for Hazardous Air Pollutants [NESHAPs] and NPDES permits) contribute to minimizing health impacts on the public. The effectiveness of these controls is verified through the use of environmental monitoring information and inspection of mitigation measures.

Baseline air emission concentrations and applicable standards for hazardous chemicals are addressed in Section 3.3.1. The baseline concentrations are estimates of the highest existing offsite concentrations and represent the highest concentrations to which members of the public could be exposed. These concentrations are in compliance with applicable guidelines and regulations. The baseline water data for assessing potential health impacts from the chemical environment are addressed in Section 3.3.8, Water Resources.

Workers are protected from workplace hazards through appropriate training, protective equipment, monitoring, materials substitution, and engineering and management controls. They are also protected by adherence to the Occupational Safety and Health Administration Process Safety Management and workplace limits, and EPA standards that limit workplace atmospheric and drinking water concentrations of potentially hazardous chemicals. DOE also requires that conditions in the workplace be as free as possible from recognized hazards that cause, or are likely to cause, illness or physical harm.

3.3.3 Socioeconomics and Environmental Justice

3.3.3.1 Socioeconomics

Statistics for the local economy, population, and housing are presented for the socioeconomics region of influence (ROI)[24], a four-county area spanning Georgia and South Carolina that includes Columbia and Richmond counties in Georgia and Aiken and Barnwell counties in South Carolina.

Table 3-9 provides residence information for the ROI. In 2014, 7,224 persons were directly employed at SRS, 87 percent (6,291 out of 7,224 persons employed at SRS) of whom reside in the ROI. Direct onsite employment accounted for approximately 3.3 percent of total employment in the ROI in 2013.

Table 3-9: Distribution of Employees by Place of Residence in the Savannah River Site Region of Influence in 2014

County	Number of Employees	Percent of Total Site Employment
Aiken	3,860	53
Barnwell	459	6
Columbia	1,152	16
Richmond	820	11
Region of Influence Total[a]	**6,291**	**87**

[a] Totals may not add due to rounding.

Source: DOE 2014a

Indirect employment generated by SRS operations has been calculated using a weighted average of RIMS II [Regional Input-Output Modeling System] direct-effect employment multipliers from the U.S. Bureau of Economic Analysis for select industries that most accurately reflect the major activities at the site. This method resulted in an estimated SRS direct-effect employment multiplier of 2.19. Therefore, the direct employment of 6,291 at SRS within the ROI would generate indirect

[24] Region of Influence (ROI) is a geographic extent that is being evaluated for a particular resource, generally limited to those geographic areas that are potentially impacted by the proposed action. The ROI will vary across different environmental aspects. For the purposes of this EA, the ROI for socioeconomics is defined as the counties where approximately 90 percent of the Savannah River Site workforce resides.

employment of 7,486, resulting in a total employment of 13,777 within the ROI, or 6.2 percent of the total employment in the ROI in 2013.

3.3.3.2 Regional Economic Characteristics

Between 2000 and 2013, the civilian labor force of the ROI increased at an average annual rate of 0.8 percent, to 237,872. At the same time, employment in the ROI increased at an average annual rate of 0.5 percent to 220,989, resulting in a 3.7 percentage point increase in the unemployment rate. Unemployment in the ROI was 7.1 percent in 2013, up from the 2000 level of 3.4 percent. Georgia and South Carolina experienced similar trends in unemployment rates, increasing 4.1 percentage points and 2.9 percentage points over the 14-year period, respectively (BLS 2014).

From 2000 to 2012, the average real per capita income of the ROI increased by approximately 2.8 percent in 2012 dollars, to $34,833. South Carolina experienced a slightly larger increase than in the ROI, increasing 4.5 percent to $35,056. The per-capita income of Georgia decreased 2.2 percent to $37,449 over the same time period (BEA 2014a). **Table 3-10** presents the per capita incomes of the ROI, Georgia, and South Carolina.

Table 3-10: Per Capita Income of the Savannah River Site Region of Influence, Georgia, and South Carolina in 2000 and 2012

Year	Savannah River Site Region of Influence		Georgia		South Carolina	
	Nominal	Real[a]	Nominal	Real[a]	Nominal	Real[a]
2000	$25,370	$33,874	$28,672	$38,282	$25,124	$33,545
2012	$34,833	$34,833	$37,449	$37,449	$35,056	$35,056

[a] Real per capita income adjusted to 2012 dollars using the Consumer Price Index for All South Urban Consumers in U.S. City Average.

Source: BEA 2014a.

In 2012, government agencies, including Federal, state and local governments, were the largest employers in the ROI, at approximately 20 percent of total employment. Retail trade was the next leading industry at approximately 11 percent of employment, followed by healthcare and social assistance at approximately 10 percent. Similar employment distributions were seen in Georgia, where the leading employment sectors were also government, retail trade and healthcare and social assistance at approximately 14 percent, 10 percent, and 9 percent, respectively. South Carolina's leading employment sectors were government, retail trade, and manufacturing at approximately 16 percent, 11 percent, and 9 percent, respectively (BEA 2014b).

3.3.3.3 Population and Housing

The 2013 population in the ROI was estimated to be 523,714 (Census 2014). From 2000 to 2013, the total population in the ROI increased at an average annual rate of approximately 1.1 percent, which was lower than the growth rate in both Georgia and South Carolina. The populations of the ROI, Georgia, and South Carolina are shown in **Table 3-11**.

Table 3-11: Total Population in the Savannah River Site Region of Influence, Georgia, and South Carolina in 2000 and 2013

Year	Savannah River Site Region of Influence	Georgia	South Carolina
2000	455,096	8,186,653	4,012,023
2013[a]	523,714	9,992,167	4,774,839

[a] 2013 data are an estimate based on the 2010 Census

Source: Census 2014

From 2000 to 2013, the number of housing units in the ROI increased at an average annual rate of 1.3 percent, to 222,174 units which was lower than the growth rate in both Georgia and South Carolina (Census 2014). **Table 3-12** shows the number of housing units in the ROI, Georgia, and South Carolina.

Table 3-12: Total Housing Units in the Savannah River Site Region of Influence, Georgia, and South Carolina in 2000 and 2013

Year	Savannah River Site Region of Influence	Georgia	South Carolina
2000	187,811	3,281,737	1,753,670
2013	222,174	4,109,896	2,158,652

Source: Census 2014

3.3.3.4 Environmental Justice

Environmental justice concerns the environmental impacts that proposed actions may have on minority and low-income populations, and whether such impacts are disproportionate to those in the population as a whole in the potentially affected area. The potentially affected area for SRS includes parts of 28 counties throughout Georgia and South Carolina that make up an area within a 50-mile (80-kilometer) radius of SRS. To be consistent with the human health analysis, the population distributions of the potentially affected area were calculated using data at the block-group level of spatial resolution from the 2010 census (Census 2011a), and were projected to the year 2020 using data from the 1990 census, the 2000 census, and the 2010 census for each of the affected counties within a 50-mile (80-kilometer) radius of SRS (Census 1990, 2001, 2011a, 2012a).

In accordance with Council on Environmental Quality (CEQ) guidance, meaningfully greater minority populations were identified where either the minority population of the affected area exceeds 50 percent, or the minority population percentage of the affected area was meaningfully greater than the minority population percentage in the general population or other appropriate unit of geographic analysis (CEQ 1997). Meaningfully greater is defined here as 20 percentage points above the population percentage in the general population. The average minority population percentage of South Carolina and Georgia for the projected 2020 population is approximately 44.6 percent and the average minority population percentage of the counties surrounding SRS is approximately 42.6 percent (DOE 2015a). Comparatively, a meaningfully greater minority population percentage relative to the general population of the state and the surrounding counties would exceed the 50 percent threshold defined by CEQ. Therefore, the lower threshold of 50 percent was used to identify areas with meaningfully greater minority populations surrounding SRS. In order to evaluate the potential impacts on populations in closer proximity to the proposed sites at SRS, additional radial distances of 5, 10, and 20 miles (8, 16, and 32 kilometers) were also analyzed.

Table 3-13 shows the composition of the ROI surrounding the proposed SRS facilities at each of these distances. No populations reside within the 5-mile (8-kilometer) radius of the facilities analyzed. The total projected population residing in the SRS ROI in 2020 would be approximately 886,276, of which 47 percent would be considered members of a minority population. Of the 580 block groups in the potentially affected area, approximately 265 (46 percent) were identified as containing meaningfully greater minority populations (DOE 2015a).

The overall composition of the projected populations within every radial distance is predominantly nonminority. The concentration of minority populations is greatest within the 50 mile (80 kilometer) radius. The Black or African American population is the largest minority group within every radial distance, constituting approximately 37 percent of the total population within 50 miles (80 kilometers). The Hispanic or Latino population constitutes about 5 to 6 percent of the total population at each radial distance (DOE 2015a).

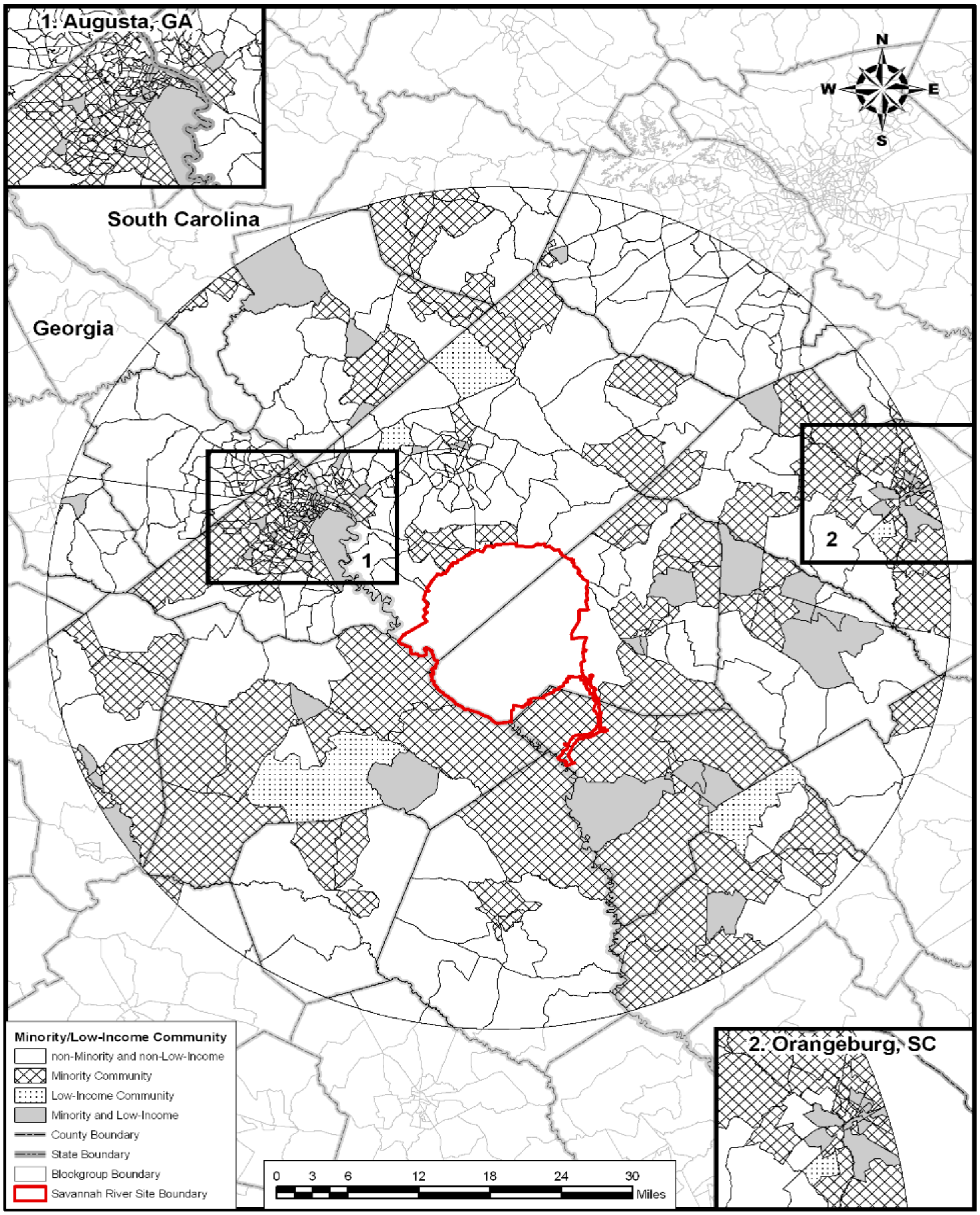
Table 3-13: Projected Populations in the Potentially Affected Area Surrounding the Savannah River Site in 2020

Population Group	10 Miles		20 Miles		50 Miles	
	Population	Percent of Total	Population	Percent of Total	Population	Percent of Total
Nonminority	4,216	60	73,173	64	472,377	53
Black or African American[a]	2,179	31	32,262	28	332,231	37
Total Hispanic[b]	413	6	5,429	5	46,107	5
American Indian or Alaska Native[a]	29	0	641	1	3,870	0
Other Minority[a]	634	9	9,034	8	77,789	9
Total Minority[a]	2,842	40	41,937	36	413,890	47
Total Population	7,058	100	115,110	100	886,267	100
Low-Income	1,347	19	20,433	18	162,157	18

[a] Includes Hispanic persons.
[b] Includes all Hispanic persons regardless of race.
Note: To convert miles to kilometers, multiply by 1.609. Totals may not equal the sum of subcategories due to rounding. The potentially affected area comprises the area within a 50-mile (80-kilometer) radius of the site.
Source: DOE 2015a

The projected low-income population (those living below the poverty threshold) living within 50 miles (80 kilometers) of SRS in 2020 was estimated to be 162,157 people (18.3 percent) (DOE 2015a). Meaningfully greater low-income populations were identified using the same methodology described for identification of minority populations. The 2010 census does not contain any data relative to income. The U.S. Census Bureau's American Community Survey (ACS) 5-year estimates are the only data set that publishes current data relative to income at the block group level of geography. Therefore, the 2006–2010 ACS 5-year estimates were used to identify low-income populations in the potentially affected area. These populations were then scaled up to be directly comparable to the projected 2020 potentially affected population. The 2006–2010 ACS 5-year estimates show the average low-income population percentage of South Carolina and Georgia is 15.9 percent (Census 2011b). Comparatively, a meaningfully greater low-income population percentage using these statistics would be 35.9 percent. Therefore, the lower threshold of 35.9 percent was used to identify areas with meaningfully greater low-income populations surrounding SRS. Of the 580 block groups that surround SRS, 80 (14 percent) contain meaningfully greater low-income populations (DOE 2015a).

Figure 3-4 displays the block groups identified as having meaningfully greater minority and low-income populations surrounding SRS. Of the 580 block groups that surround SRS, 72 (12 percent) contain both meaningfully greater minority and low-income populations.



Source: DOE 2015a

Figure 3-4: Meaningfully Greater Minority and Low-Income Populations Surrounding the Savannah River Site

3.3.4 Waste Management

Waste management includes minimization, characterization, treatment, storage, and disposal of solid and liquid waste from DOE activities. The waste is managed according to appropriate treatment, storage, and disposal technologies in compliance with applicable Federal and state statutes and DOE orders. Site wide remediation activities are conducted under a 1993 Federal Facility Agreement, a tri-party agreement between the EPA, the SCDHEC, and DOE. The Federal Facility Agreement directs the comprehensive remediation of the site and integrates cleanup requirements under the Resource Conservation and Recovery Act (RCRA) and the Comprehensive Environmental Response, Compensation, and Liability Act (SRNS 2013).

3.3.4.1 Waste Generation, Treatment, Storage, and Disposal at SRS

Table 3-14 summarizes SRS generation rates through fiscal year 2014 for waste types expected to be generated under the alternatives evaluated in this EA: low-level radioactive waste (LLW), including small quantities of LLW containing polychlorinated biphenyls; hazardous waste; nonhazardous solid waste; and construction and demolition debris. While SRS operations generate transuranic (TRU) waste and mixed low-level radioactive waste, those waste streams are not discussed here because DOE does not expect the proposed alternatives would generate those wastes.[25] Generation rates for HLW, liquid LLW, and liquid sanitary waste are discussed in the following sections. Annual volumes of liquid wastes solidified at the Z-Area saltstone facilities are, however, included in Table 3-14 because the solidified liquids are disposed of on-site as LLW.

Tables 3-15, 3-16, and **3-17,** respectively, provide summaries of current and planned treatment, storage, and disposal capabilities at SRS for the wastes addressed in this EA. These capabilities are described in the following sections by waste type. As shown in the tables, onsite treatment capacity is available for liquid HLW, solid and liquid LLW, and liquid nonhazardous waste. Onsite storage capacity is available for solid and liquid HLW, liquid LLW, and hazardous waste. Onsite disposal capacity is available for LLW and solid nonhazardous waste, including construction and demolition debris. Solid LLW may be disposed of on- or off-site while hazardous waste is disposed of offsite. Site discharge permits allow for the treatment and discharge of certain liquid LLW and nonhazardous waste effluents. Only those liquid waste streams that are treated consistent with permit requirements may be discharged to permitted outfalls. Solid nonhazardous waste and construction debris are disposed on-site.

[25] The *SRS Radioactive Waste Requirements* defines MLLW as waste containing both a radioactive component subject to the Atomic Energy Act of 1954, as amended, and a hazardous component subject to the Resource Conservation and Recovery Act (RCRA) (DOE 2014e). Some of the LLW generated under the proposed alternatives may include small quantities of polychlorinated biphenyls which are subject to the Toxic Substances Control Act rather than RCRA and may be disposed of at SRS as LLW (DOE 2014e).

Table 3-14: Waste Generation Rates at the Savannah River Site

	Savannah River Site – Total		L-Area Complex		H-Canyon in H-Area		HB-Line in H-Area		DWPF in S-Area		Z-Area Saltstone		E-Area and Hazardous/Mixed Waste Storage	
Waste Type	5-Year Average	FY 2014	5-Year Average	FY 2014	5-Year Average	FY 2014	5-Year Average	FY 2014	5-Year Average	FY 2014	5-Year Average	FY 2014	5-Year Average	FY 2014
LLW[a]	13,000	4,000	250	60	450	400	60	30	350	430	180	120	5	5
Hazardous[a]	24	77	0	0	0	0	0	0	0	0	N/A	N/A	0	0
Nonhazardous solid waste[a,b]	7,000	2,500	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
C&D debris[a, c]	70,000	35,500	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

C&D = construction and demolition; DWPF = Defense Waste Processing Facility; FY = fiscal year; LLW = low-level radioactive waste; N/A = not available;

[a] Waste generation expressed as cubic meters

[b] Sanitary waste generation is provided for all of the Savannah River Site (information by individual area is not available). Waste sent to the recycle facility and Three Rivers Regional Landfill is measured by weight with volume estimated at 1 metric ton per cubic meter (1,690 pounds per cubic yard).

[c] C&D debris generation is provided for all of the Savannah River Site (information by individual area is not available). C&D landfill waste volume is based on truck volumes received. About 36 percent of the reported waste mass/estimated volume is sent to the recycling facility and not disposed of in the C&D landfill. Waste generation does not include waste-like materials recovered through salvage and excess property operations, or materials recovered through construction services.

Note: To convert cubic meters to cubic feet, multiply by 35.314.

Source: Maxted 2014.

Table 3-15: Waste Treatment Capabilities at the Savannah River Site

Facility Name	Capacity	Status	Waste Type		
			High-Level Radioactive	LLW	Nonhazardous
Treatment Facility					
Defense Waste Processing Facility	275 canisters per year nominal[a]	Operating	X		
Tank Farm Evaporators	2H-Evaporator: 810,000 liters per week[b]; 2F and 3H-Evaporators: 2.1 million liters per week total	Operating		X	
Saltstone Waste Processing Facility	34 million liters per year, maximum rate	Planned for 2018	X		
Interim processing of salt waste	15 liters per minute	Operating	X		
F- and H-Areas Effluent Treatment Project	590 million liters per year	Operating		X	
Z-Area Saltstone Production Facility	28,400 cubic meters per year	Operating		X	
Central Sanitary Wastewater Treatment Facility	1.5 billion liters per year	Operating			X

[a] For sludge waste processing.

[b] Expected average annual rate of treatment of the DWPF recycle. The 2H-Evaporator only treats the DWPF recycle. All evaporators are assumed to operate at 50 percent utility.

[c] The interim processing facility, which will ultimately be replaced by the SWPF, processes salt waste from the high-level radioactive waste tanks to separate the higher activity fraction of the waste (to be sent to the DWPF for vitrification) from the lower activity fraction of the waste (to be sent to the Z-Area saltstone facilities for disposal).

Note: To convert cubic meters to cubic feet, multiply by 35.315; to convert liters to gallons, multiply by 0.26417.

Source: DOE 1999, 2015; SRR 2014a, 2014b; WSRC 2006a, 2007a, 2007b.

Table 3-16: Waste Storage Capabilities at the Savannah River Site

Facility Name	Capacity	Status	Waste Type		
			High-Level Radioactive	Low-Level Radioactive	Hazardous
Storage Facility					
High-Level Liquid Radioactive Waste Tank Farms	8.7 million liters [a]	Operating	X		
Glass Waste Storage Buildings	4,590 canisters in two existing buildings	Operating	X		
Failed Equipment Storage Vaults (Defense Waste Processing Facility)	2 exist, space allocated for 12 more vaults	Operating	X		
Transuranic Waste Storage Pads[b]	13,200 cubic meters	Operating			X
Solvent Storage Tanks S33–S36 in H-Area	105,000 liters per tank[c]	Operating		X	

[a] Operational working capacity remaining in the F- and H-Area tank farms that does not include six tanks in F-Area that have been closed or space in other tanks that may not be viable for storage or is maintained for safety reasons. Currently, 37 million gallons (140 million liters) of high-level radioactive waste are stored in 45 underground storage tanks.

[b] The Transuranic Pads are permitted to accept hazardous waste for storage.

[c] Operating capacity.

Note: There are no dedicated low-level radioactive waste storage facilities. To convert cubic meters to cubic feet, multiply by 35.315; to convert liters to gallons, multiply by 0.26417.

Source: DOE 1999; DOE 2015a; SRR 2014a, 2014b; WSRC 2007a

Table 3-17: Waste Disposal Capabilities at the Savannah River Site

Facility Name	Capacity	Status	Waste Type	
			Low-Level Radioactive	Nonhazardous
Disposal Facility				
Intermediate-Level Low-Level Radioactive Waste Vaults[a]	4,300 cubic meters per vault	Operating	X	
Low-Activity Low-Level Radioactive Waste Vaults[a]	30,500 cubic meters per vault	Operating	X	
Low-level radioactive waste disposal facility slit trenches [a]	360,000 cubic meters	Operating	X	
Low-level radioactive waste disposal facility engineered trenches[a]	140,000 cubic meters	Operating	X	
Z-Area Saltstone Disposal Facility	Current circular disposal vaults each hold about 11 million liters of grouted waste; future circular disposal vaults will each hold about 114 million liters of grouted waste.	Operating	X	
Three Rivers Regional Landfill[b]	4.2 million cubic meters per year (permitted)	Operating		X
Construction and demolition debris landfill	2.47 million cubic yards total permitted capacity	Operating		X

[a] As of October 2014, the estimated unused disposal capacity remaining is approximately 21,300 cubic meters for the Low-Activity Low-Level Radioactive Waste Vaults, 180,000 cubic meters for the slit trenches, and 75,000 cubic meters for the engineered trenches. The Intermediate-Level Low-Level Radioactive Waste Vaults are used for disposal of waste containing larger quantities of isotopes such as tritium and waste having surface radiation levels exceeding 100 millirem per hour.

[b] The Three Rivers Regional Landfill is permitted to annually receive up to 500,000 metric tons of compacted solid waste. Assuming a pre-compaction density of 200 pounds per cubic yard, approximately 4.2 million cubic meters of pre-compacted waste can be annually disposed of at the landfill.

Note: Only low-level radioactive waste and nonhazardous waste are disposed of at SRS. To convert cubic meters to cubic feet, multiply by 35.315; cubic yards to cubic meters, multiply by 0.76456; liters to cubic meters, multiply by 0.26417.

Source: DOE 1999; DOE 2015a; Maxted 2014, SRR 2013; WSRC 2007a.

3.3.4.2 High-Level Radioactive Waste

The F- and H-Area tank farms have received over 150 million gallons (570 million liters) of HLW liquid waste from SRS operations (SRR 2014b). Currently, approximately 37 million gallons (140 million liters) of waste containing about 287 million curies of radioactivity are stored in 45 underground tanks (SRR 2014a, 2014b). Approximately 2.3 million gallons (8.7 million liters) of operational working capacity remains in the F- and H-Area tank farms (SRR 2014b).

DOE is using a process involving deliquification, dissolution, and adjustment to treat certain salt waste, with additional processing of salt waste using the Actinide Removal Process and Modular Caustic Side Solvent Extraction Unit (SRNS 2009). The treatment process results in a high-activity, low-volume HLW liquid waste stream that is vitrified at DWPF, and a low-activity, high-volume LLW liquid waste stream (salt solution) that is disposed of onsite after processing at the Z-Area saltstone facilities. After completion of the Salt Waste Processing Facility, expected to become operational in 2018 (SRR 2014a), additional salt waste treatment capacity will be available.

DWPF was constructed to solidify liquid HLW stored in the F- and H-Area tank farms into a vitrified form for eventual geologic disposal, which would then allow the HLW tanks to be closed. DWPF began operating in March 1996, and is projected to complete vitrification of the HLW in the F- and H-Area tank farms by 2039 (SRR 2014b). Operations consist of mixing a sand-like borosilicate glass (called frit) with the waste, melting the mixture, and pouring it into stainless steel canisters to cool and harden. Each canister is 10 feet (3 meters) tall and 2 feet (0.6 meters) in diameter and has a filled weight of about 5,000 pounds (2,300 kilograms). Filled canisters are taken from DWPF to the adjacent glass waste storage facilities. The estimated storage capacity of the existing two storage buildings is approximately 4,590 canisters (SRR 2014a). DOE is planning to develop additional storage capacity. The canisters will remain in safe, secure storage pending decisions on a long-term solution for management of HLW and spent nuclear fuel.[26] Through December 31, 2013, 3,754 canisters of waste containing about 52 million curies had been poured at DWPF (SRR 2014b).

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[26] DOE terminated the program for a geologic repository for spent nuclear fuel and HLW at Yucca Mountain, in Nevada. Notwithstanding the decision to terminate the Yucca Mountain program, DOE remains committed to meeting its obligations to manage and ultimately dispose of spent nuclear fuel and HLW. DOE established the Blue Ribbon Commission on America's Nuclear Future to conduct a comprehensive review and evaluate alternative approaches for meeting these obligations. The Commission report to the Secretary of Energy of January 26, 2012 (BRCANF 2012) provided a strong foundation for the development of the Administration's January 2013 Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste (DOE 2013b). This Strategy provides a framework for moving toward a sustainable program to deploy an integrated system capable of transporting, storing, and disposing of spent nuclear fuel and high-level radioactive waste from civilian nuclear power generation, defense, national security, and other activities. The link to the strategy is http://energy.gov/downloads/strategy-management-and-disposal-used-nuclear-fuel-and-high-level-radioactive-waste. Full implementation of this Strategy will require legislation.

3.3.4.3 Low-Level Radioactive Waste

Both liquid and solid LLW are treated at SRS. Most aqueous LLW streams are sent to the F- and H-Area Effluent Treatment Project (formerly called the Effluent Treatment Facility) and treated by pH adjustment, organic removal, reverse osmosis, and ion exchange to remove chemical and radioactive contaminants other than tritium. This facility is designed to process 100,000 to 250,000 gallons (380,000 to 950,000 liters) of low-level radioactive wastewater daily. The maximum permitted facility capacity is 430,000 gallons (1.6 million liters) per day, or about 160 million gallons (590 million liters) per year. Actual processing is approximately 20 million gallons (76 million liters) of wastewater per year, or 55,000 gallons (210,000 liters) per day (WSRC 2006a, 2006b, 2007a). After treatment, the effluent is discharged to Upper Three Runs through an NPDES permitted outfall. The treatment residuals are concentrated by evaporation and stored in the H-Area tank farm for eventual treatment in the Z-Area saltstone facilities where wastes are immobilized with grout for onsite disposal (SRR 2012).

Most solid LLW is disposed of at SRS in engineered trenches and slit trenches. As of October 2014, about 98,000 cubic yards (75,000 cubic meters) of disposal space remained in the engineered trenches and about 235,000 cubic yards (180,000 cubic meters) of disposal space remained in the slit trenches. Concrete vaults located in E-Area are used to dispose of the higher radioactive fraction of the LLW generated at SRS (Maxted 2014). Although most solid LLW is disposed of on site at SRS, some LLW is shipped off site for disposal at DOE’s Nevada National Security Site and commercial facilities (SRNS 2009).

Low-activity liquid wastes including liquid waste from the Effluent Treatment Project and salt solution separated from HLW are processed and disposed in the saltstone facilities in Z-Area. Saltstone, a solidified grout formed by mixing liquid waste with cement, fly ash, and furnace slag, is produced in the Saltstone Production Facility. From there, the saltstone slurry is mechanically pumped to the Saltstone Disposal Units for disposal as LLW (SRR 2012).

3.3.4.4 Hazardous Waste

Hazardous waste is nonradioactive waste that SCDHEC regulates under RCRA and corresponding state regulations. Hazardous waste is accumulated at the generating location or stored in U.S. Department of Transportation (DOT)-approved containers in E-Area. Hazardous waste is shipped off site to commercial RCRA-permitted treatment and disposal facilities using DOT-certified transporters (DOE 1999). DOE also recycles, reuses, or recovers certain hazardous wastes such as metals, excess chemicals, solvents, and chlorofluorocarbons (DOE 2002).

3.3.4.5 Nonhazardous Waste

Solid nonhazardous waste is sent to the Three Rivers Regional Landfill, which is located within the SRS site boundary (DOE 2002) and serves as a regional municipal landfill for Aiken, Allendale, Bamberg, Barnwell, Calhoun, Edgefield, McCormick, Orangeburg, and Saluda Counties. The Three Rivers Regional Landfill has a 300-acre (120-hectare) footprint with a remaining capacity in excess of 38 million cubic yards (29 million cubic meters) of waste as of 2014 (TRSWA 2014). Although the landfill is permitted to annually receive up to 550,000 tons (500,000 metric tons) of nonhazardous solid waste (DOE 2015a), it typically annually receives about 250,000 tons (230,000 metric tons) of waste (TRSWA 2014). Construction and demolition debris from SRS activities is disposed of in an onsite landfill (DOE 2015a).

Liquid nonhazardous waste (sanitary wastewater) is collected and treated at the Central Sanitary Wastewater Treatment Facility prior to NPDES-permitted outfalls. The Central Sanitary Wastewater Treatment Facility has a design capacity to treat up to 383 million gallons (1.5 billion liters) per year (DOE 2015a).

3.3.5 Transportation and Traffic

The Savannah River Site is serviced by a system of Interstate, U.S. and state highways, and railroads. SRS is managed as a controlled area with limited public access.

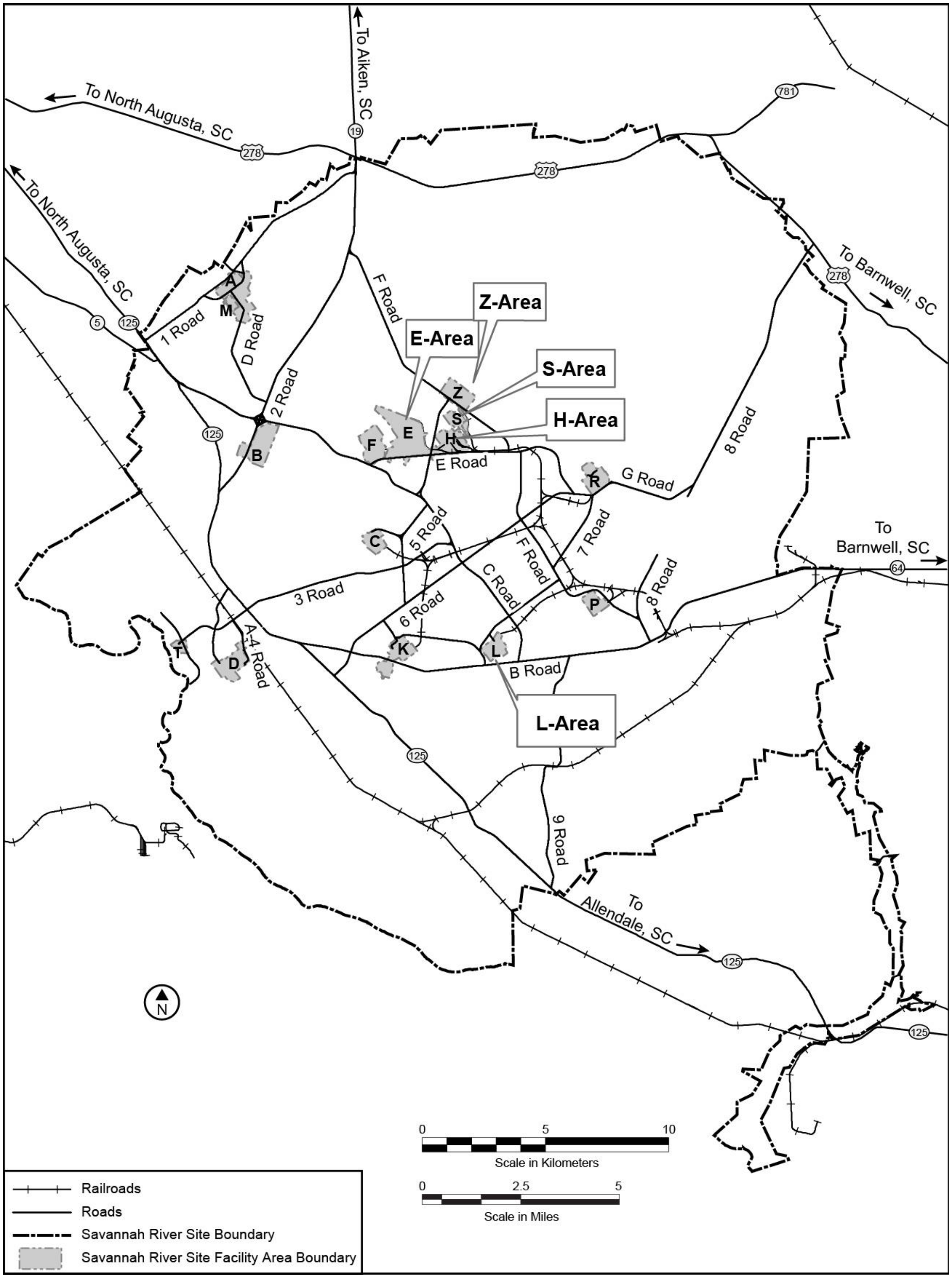
The regional transportation networks provide service to SRS employees residing in South Carolina and Georgia. Vehicular access to SRS is provided from South Carolina State Highways 19, 64, 125, 781, and U.S. Highway 278. Commuter traffic between SRS and Georgia crosses the Savannah River primarily on I–20 and I–520 and primary arteries Routes 28 and 1 and Business Route 25 to the north of SRS.

There are several major road improvement projects in the area. In Augusta, Georgia, the River Watch Extension project began in February 2014. This project includes: widening the road to four lanes, building a bridge over the CSX Railroad, and adding sidewalks and bicycle lanes. The project will also extend the road by 2.5 miles (4 kilometers) from Baston Road to Washington Road (City of Augusta 2014). Other improvements are being performed on Wrightsboro Road, SR 232/Columbia Road, SR 56/Mike Padgett Highway, and the Augusta Canal Multi-Use Trail (City of Augusta 2014).

Within SRS, there are approximately 130 miles (209 kilometers) of primary and 1,100 miles (1,770 kilometers) of secondary roads (DOE 2005a). The primary SRS roadways are in good condition, and are typically wide, firm shoulder border roads that are either straight or have wide gradual turns. Intersections are well marked for both traffic and safety identification.

In addition to the vehicular roadways, railroads are used for transporting large volumes or oversized loads of materials or supplies (DOE 2005a). As shown in **Figure 3-5**, travel between facilities in L-, E-, and H-Areas evaluated in this Draft EA can be accomplished by both surface roads and railroads.

Rail service in the region is provided by the Norfolk Southern Corporation and CSX Transportation. Rail access to SRS is provided by the Robbins Station on the CSX Transportation line (DOE 1999). Within SRS, there are approximately 32 miles (51 kilometers) of railroad track (DOE 2015a). The railroad tracks are well maintained, and the rails and cross lines are in good condition. The Savanah River rail classification yard is east of P area. This facility sorts and redirects railroad cars. The railroads support delivery of foreign and domestic research reactor spent nuclear fuel shipments, delivery of construction materials for new projects, and movement of nuclear materials and equipment on site (DOE 2005a).



Source: DOE 2015a

Figure 3-5: Savannah River Site Transportation Infrastructure

3.3.6 Land Resources

3.3.6.1 Land Use

Predominant regional land uses in the vicinity of SRS include urban, residential, industrial, agricultural, and recreational. SRS is bordered mostly by forest and agricultural land, with limited urban and residential development. The nearest residences are located to the west, north, and northeast, some within 200 feet (61 meters) of the SRS boundary (DOE 2015a). Farming is diversified throughout Aiken, Allendale, and Barnwell Counties and includes such crops as corn, hay, peanuts, cotton, and winter wheat (USDA 2011). Industrial areas are also present within 25 miles (40 kilometers) of the site; industrial facilities include textile mills, polystyrene foam and paper plants, chemical processing plants, the Barnwell LLW facility, and a commercial nuclear power plant. Open water and nonforested wetlands occur along the Savannah River Valley. Recreational areas within 50 miles (80 kilometers) of SRS include Sumter National Forest, Santee National Wildlife Refuge, and Clark's Hill/Strom Thurmond Reservoir. State, county, and local parks include Redcliffe Plantation, Rivers Bridge, Barnwell State Park, and the Aiken State Natural Area in South Carolina, and Mistletoe State Park in Georgia. The Crackerneck Wildlife Management Area occupies a portion of SRS along the Savannah River and is open to the public for hunting and fishing at certain times of the year (DOE 2015a).

Land use at SRS can be classified into three major categories: forest/undeveloped, water/wetlands, and developed facilities. Open fields and pine and hardwood forests make up 73 percent of the site; 22 percent is wetlands, streams, and two lakes (DOE 2015a). Production and support areas, roads, and utility corridors account for the remaining 5 percent of the land area.

The U.S. Forest Service, under an interagency agreement with the DOE, manages timber production on about 149,000 acres (60,300 hectares) (USFS-Savannah River 2004). Public hunts for white-tailed deer (*Odocoileus virginianus*), feral hogs (*Sus scrofa*), wild turkeys (*Meleagris gallopavo*), and coyote (*Canis latrans*) are allowed on site at specified times.

Soil map units that meet the requirements for prime farmland soils exist on the site. However, the Natural Resources Conservation Service (NRCS) of the U.S. Department of Agriculture does not identify these as prime farmlands because the land is not available for agricultural production (DOE 2015a).

The site has been divided into six management areas based on existing biological and physical conditions, operations capability, and suitability for mission objectives: the 38,444-acre (15,558-hectare) Industrial Core Management Area, the 87,200-acre (35,289-hectare) Red-cockaded Woodpecker Management Area, the 47,100-acre (19,061-hectare) Supplemental Red-cockaded Woodpecker Management Area, the 10,400 acres (4,209 hectares) Crackerneck Wildlife Management Area and Ecological Reserve managed by the South Carolina Department of Natural Resources, the 10,000-acre (4,047-hectare) Savannah River Swamp, and 4,400-acre (1,780-hectare) Lower Three Runs Corridor Management Area. The 38,444-acre (15,558-hectare) Industrial Core Management Area contains the major SRS facilities.

In 1972, all of SRS was designated as a National Environmental Research Park. The purpose of the National Environmental Research Park is to conduct research and education activities to assess and document environmental effects associated with energy and weapons material production, explore methods for eliminating or minimizing adverse effects of energy development and nuclear materials on the environment, train people in ecological and environmental sciences, and educate the public (SREL 2010).

DOE has prepared a number of documents addressing the future of SRS, including the Savannah River Site End State Vision report (DOE 2005a) and the Savannah River Site Comprehensive Plan/Ten Year Plan, FY 2015 - 2024 (SRNS 2014c). SRS recently updated Appendix H of the Federal Facility Agreement to show that the Environmental Management Cleanup Project and mission will be complete by 2065. The National Nuclear Security Administration nuclear industrial missions will continue. SRS is a site with an enduring mission and is not a closure site; thus, SRS land will be federally owned, controlled, and maintained in perpetuity (DOE 2005a).

3.3.6.2 Visual Resources

The dominant viewshed in the vicinity of SRS consists mainly of agricultural land and forest, with some limited residential and industrial areas. The SRS landscape is characterized by wetlands and upland hills. Vegetation comprises bottomland hardwood forests, scrub oak and pine forests, and forested wetlands. Facilities are scattered throughout SRS and are brightly lit at night. These facilities are generally not visible off site, as views are limited by rolling terrain, normally hazy atmospheric conditions, and heavy vegetation. The only areas visually impacted by the DOE facilities are those within the view corridors of State Highway 125 and U.S. Highway 278 (DOE 2015a).

The developed areas and utility corridors (transmission lines and aboveground pipelines) of SRS are consistent with a Visual Resource Management Class IV designation. The remainder of SRS is consistent with a Visual Resource Management Class II or Class III designation. Management activities within Class II and Class III areas may be seen, but do not dominate the view; management activities in Class IV areas dominate the view and are the focus of viewer attention (DOI 1986).

3.3.7 Geology and Soils

3.3.7.1 Geology

SRS is situated primarily on the Aiken Plateau of the Upper Atlantic Coastal Plain physiographic region, approximately 25 miles (40 km) southeast of the Fall Line that separates the Atlantic Coastal Plan from the Piedmont physiographic province. The Aiken Plateau is highly dissected and characterized by broad, flat areas between streams and narrow, steep-sided valleys. The Aiken Plateau ranges in elevation from 250 to 400 feet (76 to 122 meters) above mean sea level. The alluvial terraces of the Savannah River occur below 250 feet (76 meters) above mean sea level (DOE 2011c).

Geologic faults have been identified on SRS, but none of these faults are considered to be capable, meaning that none of these faults, or associated faults, has moved at or near the surface within the past 35,000 years (DOE 2011c). The only known faults capable of producing an earthquake within a 200-mile (320-kilometer) radius of SRS are within the Charleston seismic zone (located approximately 70 miles [110 kilometers] southeast of SRS) (NRC 2005; USGS 2014a).

The Charleston, South Carolina, earthquake of 1886 (estimated Richter scale magnitude of 6.8) is the most damaging earthquake known to have occurred in the southeastern United States and one of the largest historic shocks in eastern North America (DOE 2015a). At SRS, this earthquake had an estimated Richter scale magnitude ranging from 6.5 to 7.5. The SRS area experienced an estimated peak ground acceleration of 0.10 *g* (one-tenth the acceleration of gravity) during this event (NRC 2005).

Earthquake-produced ground motion is expressed in units of percent *g* (force of acceleration relative to that of Earth's gravity). The latest probabilistic peak (horizontal) ground acceleration (PGA) data from the U.S. Geological Survey were used to indicate seismic hazard. The PGA values cited are based on a 2 percent probability of exceedance in 50 years (See USGS 2014b). This corresponds to an annual occurrence probability of about 1 in 2,500. At the center of SRS, the calculated PGA is approximately 0.17 *g* (DOE 2015a). Most of the PGA is related to the proximity of SRS to the Charleston seismic zone and not from locally generated earthquakes. Earthquakes capable of producing structural damage are not likely to originate in the vicinity of SRS (DOE 2015a).

The loosely consolidated Atlantic Coastal Plain sediments are located above bedrock that consists of Paleozoic-age metamorphic and igneous rock (e.g., granite) and Triassic-age sedimentary rock (e.g., siltstone) of the Dunbarton Basin (NRC 2005). The Atlantic Coastal Plain sediments consist of layers of sandy clays and clayey sands, along with occasional beds of clays, silts, sands, gravels, and carbonate that dip gently and thicken to the southeast from near zero at the fall line to about 4,000 feet (1,219 meters) at the South Carolina coast (NRC 2005; WSRC 2006c, 2006d). The Atlantic Coastal Plain sediments at SRS are approximately 600 to 1,400 feet (183 to 427 meters) thick (DOE 2015a).

The Atlantic Coastal Plain sedimentary sequence near the center of SRS consists of about 700 feet (213 meters) of late Cretaceous quartz sand, pebbly sand, and kaolinitic clay, overlain by about 60 feet (18 meters) of Paleocene clayey and silty quartz sand, glauconitic sand, and silt. (DOE, 2015) The Paleocene beds are overlain by about 350 feet (107 meters) of Eocene quartz sand, glauconitic quartz sand, clay, and limestone grading into calcareous sand, silt, and clay. In places, especially at higher elevations, the sequence is capped by deposits of pebbly and clayey sand, conglomerate, and clay from the Miocene or Oligocene era (DOE 2015a). The sediment, comprising layers of sand, muddy sand, and clay with subordinate calcareous sediments, rests on crystalline and sedimentary basement rock. Water flows easily through the sand layers, but is slowed by less-permeable clay beds, creating a complex system of aquifers (DOE 2015a). These aquifers are discussed in Section 3.3.8.2, Groundwater.

3.3.7.2 Soils

The NRCS identifies 28 soil series occurring on SRS. These soil series are grouped into seven broad soil-association groups (DOE 2015a). Generally, sandy soils occupy the uplands and ridges, and loamy-clayey soils occupy the stream terraces and floodplains (CSRACT 2007). The Fuquay–Blanton–Dothan Soil Association consists of nearly level to sloping, well-drained soils on the broad upland ridges, including most undisturbed soils near E-, F-, H-, K-, and S-Areas. This association covers approximately 47 percent of SRS and is composed of about 20 percent Fuquay soils, 20 percent Blanton soils, 12 percent Dothan soils, and 48 percent other soils (WSRC 2006d). Fuquay and Dothan soils are well drained, and Blanton soils are somewhat excessively drained.

These soils have moderately thick to thick sandy surface and subsurface layers and loamy subsoil. Most of these soils are suited for cultivated crops, timber production, sanitary facilities, and building sites (WSRC 2006d). The soils at SRS are considered acceptable for standard construction techniques (DOE 2015a).

3.3.8 Water Resources

3.3.8.1 Surface Water

Surface water drainage in the region is dominated by the Savannah River, which forms the western boundary of SRS. The Savannah River receives drainage from five major tributaries which originate on or drain through SRS. These tributaries are Upper Three Runs, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs. No streams or tributaries at SRS are federally designated Wild and Scenic Rivers or state designated Scenic Rivers (NRC 2005; DOE 2015a). Detailed descriptions of SRS surface water hydrology can be found in the SRS Ecology Environmental Information Document (WSRC 2006d).

The Savannah River is classified by SCDHEC as freshwater that is suitable for primary and secondary contact recreation, drinking after appropriate treatment, balanced native aquatic species development, and industrial and agricultural purposes. This same use classification is applicable to the five tributaries which originate on or drain through SRS (WSRC 2006d). No SRS facilities are located within the 100-year floodplain. Probabilities of flooding in E-, H-, K-, and S-Areas are significantly less than 0.00001 per year (DOE 2015a).

SRS has five active SCDHEC NPDES permits and one no-discharge permit for land application of biosolids (SRNS 2014b). The Biomass Cogeneration Facility operated by Ameresco Federal Solutions, Inc. also maintains an industrial wastewater discharge permit which is independent of SRS's permits, and is reported separately (SRNS 2014b).

Twenty-nine NPDES-permitted industrial wastewater outfalls across SRS are monitored on a monthly basis. For each outfall, physical, chemical, and biological parameters are determined and reported to SCDHEC in SRS monthly discharge monitoring reports, as required by the permit. Annually, SRS reports more than 1,400 measurements. In 2013, the SRS NPDES program maintained a greater than 99 percent compliance rate. SRS had three permit limit exceptions during 2013, and received two notices of (SRNS 2014b). Details of the SRS NPDES limit exceptions may be found in the *SRS Environmental Report for 2013* (SRNS 2014b).

3.3.8.2 Groundwater

Groundwater velocities at SRS range from several inches to several feet per year in aquitards and from tens to hundreds of feet per year in aquifers (SRNS 2014b). This EA incorporates the groundwater system naming conventions used in the *Final SPD SEIS* (DOE 2015a). The SRS groundwater flow system is characterized by four major aquifers separated by confining units. The uppermost aquifer is referred to as the "water table aquifer." It is supported by the leaky "Green Clay" Aquitard, which confines the Congaree Aquifer. Below the Congaree Aquifer is the leaky Ellenton Aquitard, which confines the Cretaceous Aquifer, also known as the Tuscaloosa Aquifer. In general, groundwater in the water table aquifer flows downward to the Congaree Aquifer or discharges to nearby streams. Flow in the Congaree Aquifer is downward to the Cretaceous Aquifer or horizontal to stream discharge or the Savannah River, depending on the location within

SRS (DOE 2015a). Other groundwater hydrostratigraphic unit classification systems applicable to SRS are presented in the *Savannah River Site Environmental Report for 2010* (SRNS 2011). The Cretaceous Aquifer is an important water resource for the SRS region. Groundwater withdrawn in and around SRS is used extensively for domestic, industrial, and municipal purposes (DOE 2015a).

All aquifers are defined by the *South Carolina Pollution Control Act* (SC Code § 48-1-10 et seq.) as potential sources of drinking water. None of these aquifers, however, is designated as a sole-source aquifer. A sole-source aquifer is defined as an aquifer that supplies at least 50 percent of the drinking water to the area above the aquifer (EPA 2011). These areas can have no other water supply capable of physically, legally, or economically providing drinking water to local populations (NRC 2005).

Drinking water for SRS is supplied by seven regulated water supply systems, all of which utilize groundwater sources. The SRS groundwater withdrawal network includes 8 domestic water wells and approximately 32 process water wells. Samples are collected and analyzed by SRS and SCDHEC to ensure that water systems meet SCDHEC and U.S. Environmental Protection Agency (EPA) bacteriological and chemical drinking water quality standards. All drinking water samples collected and analyzed by SRS and SCDHEC met the SCDHEC and EPA bacteriological and chemical drinking-water quality standards in 2013 (SRNS 2014b).

There has been a major decline in withdrawals since annual reporting of SRS groundwater usage began in 1983. Groundwater withdrawals were reduced by more than two-thirds between the early 1980s and 2010. Total annual water use was reduced by approximately 22 percent between 2008 and 2010 (from 2.3 billion gallons [8.7 billion liters] to 1.8 billion gallons [6.8 billion liters]). Facility shutdowns, site population reductions, and water supply system upgrades and consolidation have measurably reduced SRS water use demands (SRNS 2014b).

To meet state and Federal laws and regulations, extensive groundwater monitoring is conducted annually around SRS waste sites and operating facilities, using approximately 2,000 monitoring wells (SRNS 2014b). Major contaminants include volatile organic compounds, metals, and radionuclides (SRNS 2014b). Groundwater contamination sites are primarily located in proximity to closed reactor facilities (C-, K-, L-, P-, and R-Areas), the General Separations Area (F- and H-Areas), and the waste management areas (E-, S-, and Z-Areas) (DOE 2015a). For the reactor facilities, tritium and trichloroethylene are the primary contaminants identified in groundwater plumes; concentrations of other radionuclides and organics and metals are also present. The contamination associated with the historic operations at the General Separations Area and waste management areas include smaller, frequently overlapping groundwater plumes that include trichloroethylene and tetrachloroethylene, radionuclides, metals, and other constituents (SRNS 2014b).

The water table aquifer is contaminated with solvents, metals, and low levels of radionuclides at several SRS sites and facilities. The Cretaceous Aquifer is generally unaffected except for an area near A-Area, where trichloroethylene has been reported. Trichloroethylene has also been reported in A- and M-Areas in the Congaree Aquifer. Tritium has been reported in the Congaree Aquifer in the General Separations Area, which includes F- and H-Areas. Groundwater eventually discharges into onsite streams or the Savannah River; groundwater contamination has not been detected beyond SRS boundaries (DOE 2015a).

3.3.9 Ecological Resources

3.3.9.1 Aquatic Habitat and Wetlands

Over 20 percent of SRS surface area is covered by water, including wetlands, bottomland hardwoods, cypress-tupelo swamp forests, two large cooling water reservoirs, creeks and streams, and 299 isolated upland Carolina bays and wetland depressions (DOE, 2015). There are more than 50 manmade impoundments throughout the site that support fish populations. Carolina bays, a type of wetland unique to the southeastern United States, are natural shallow depressions which can range from lakes to shallow marshes, herbaceous bogs, shrub bogs, or swamp forests. Among the 299 known or suspect Carolina bays found throughout SRS, fewer than 20 have permanent fish populations. Although fishing in SRS surface waters is prohibited, the contiguous Savannah River possesses both sport and commercial fisheries (DOE 1982). SRS wetlands, which are associated with floodplains, streams, Carolina bays, and impoundments, include vegetation such as bottomland hardwood, cypress-tupelo, emergent vegetation and swamp forest (Davis and Janecek 1997).

3.3.9.2 Terrestrial

SRS's terrestrial habitat is primarily forestland. Forested cover types at SRS include bottomland hardwood, pine forest, mixed forest, and forested wetland. Nonforested cover types include scrub shrub, emergent wetland, industrial, grassland, clearcut, and bare soil/borrow pit. Approximately 90 percent of the land cover at SRS is bottomland hardwood forests, pine forests, and mixed forests (DOE 2015a; WSRC 2006d). The biodiversity within SRS is extensive due to the variety of plant communities and the mild climate. Scientists have documented the occurrence of 1,322 plant species from 151 taxonomic families on SRS. Animal species known to inhabit SRS include 55 species of mammals, 255 species of birds, 60 species of reptiles, and 44 species of amphibians.

Common species include the eastern box turtle (*Terrapene carolina*), Carolina chickadee (*Poecile carolinensis*), common crow (*Corvus brachyrhynchos*), eastern cottontail (*Sylvilagus floridanus*), and gray fox (*Urocyon cinereoargenteus*). Game animals include a number of species, two of which, the white-tailed deer (*Odocoileus virginianus*), and feral hogs (*Sus scrofa*), are hunted on the site. Raptors, such as the Cooper's hawk (*Accipiter cooperii*) and the black vulture (*Coragyps atratus*), and carnivores, such as the gray fox, are ecologically important groups at SRS (DOE 2015a). Ecological resources at SRS are discussed in detail in the SRS Ecology Environmental Information Document (WSRC 2006d).

3.3.9.3 Threatened and Endangered Species

Five species afforded protection under the Federal Endangered Species Act (16 USC 1531 *et seq*.) are found on SRS: the wood stork (*Mycteria americana*), red cockaded woodpecker (*Picoides borealis*), shortnose sturgeon (*Acipenser brevirostrum*), smooth purple coneflower (*Echinacea laevigata*), and pondberry (*Lindera melissifolia*) (WSRC 2006d).

Although the bald eagle has been de-listed from the Endangered Species Act, it is still protected by the Migratory Bird Treaty Act (16 USC 703 *et seq*.) and the Bald and Golden Eagle Protection Act (16 USC 668-668d). SRS has a small breeding population of bald eagles (DOE 2011c). There are two established nesting sites on SRS: the Pen Branch site located west of L-Lake and the Eagle Bay site located in a cypress wetland south of Par Pond. Each nesting site is surrounded by a 6,560-

foot (2,000-meter)-wide buffer zone, with access restrictions from September 15 through June 1. At SRS, breeding eagles typically begin nest building in late fall or early winter. Chicks typically fledge and leave the nest by late spring (DOE 2011c).

Additional descriptive information on threatened and endangered species and other species found on SRS can be found in the Biological Evaluation (BE) that accompanies the EA for the Proposed Use of Savannah River Site Lands for Military Training (DOE 2011c).

3.3.10 Cultural and Paleontological Resources

Through a cooperative agreement, DOE and the South Carolina Institute of Archaeology and Anthropology (University of South Carolina) conduct the Savannah River Archaeological Research Program to provide services required by Federal law (including the National Historic Preservation Act [16 USC 470 *et seq.*]) for the protection and management of archaeological, cultural, and historical resources. To facilitate the management of these resources, SRS is divided into three zones based on an area’s potential for containing sites of archaeological, cultural, or historical significance (SRARP 1989). Zones 1, 2, and 3 represent areas possessing high, moderate, and low potential (respectively) for significant archaeological or historical resources. High priority sites are typically located on elevated areas or bluffs adjacent to stream corridors and other wetlands.

Systematic surveys for archeological (historic and prehistoric) resources have been conducted on 35 percent of the SRS area available for survey, resulting in the identification and inventory of 1,930 sites (SRARP 2012). Although most of these sites have not been formally evaluated for eligibility for listing in the National Register of Historic Places (NRHP), 67 sites have been identified as potentially eligible.

Prehistoric resources are physical properties that remain from human activities that predate written records (DOE 2015a). In general terms, prehistoric sites on SRS consist of village sites, base camps, limited-activity sites, quarries and workshops (NRC 2005).

Historic resources consist of physical properties that postdate the existence of written records. In the United States, historic resources are generally considered to be those that date no earlier than 1492 (DOE 2015a). SRS is an exceptionally important historic resource that provides information about our nation’s twentieth-century Cold War history (DOE 2015a).

American Indian resources are sites, areas, and materials important to American Indians for religious or heritage reasons. In addition, cultural values are placed on natural resources, such as plants, that have multiple purposes within various American Indian groups. Of primary concern are concepts of sacred space that create the potential for land use conflicts (DOE 2015a).

Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geological age (DOE 2015a). Paleontological materials from the SRS area date largely from the Eocene Age (54 to 39 million years ago) and include fossilized plants, invertebrate fossils, giant oysters (*Crassostrea gigantissima*), other mollusks, and bryozoa. With the exception of the giant oysters, all other fossils are fairly widespread and common; therefore, the assemblages have low research potential or scientific value (DOE 2015a).

3.3.11 Infrastructure

Site infrastructure includes those basic resources and services required to support planned construction and operations activities and the continued operations of existing facilities. For the purposes of this EA, infrastructure is defined as, electricity, fuel, water, and sewage. **Table 3-18** describes the SRS infrastructure. SRS's electricity, water, and wastewater systems are designed to support a site population of approximately 20,000 persons. In 2014, 7,224 persons were directly employed at SRS; accordingly this infrastructure possesses excess capacity.

Table 3-18: SRS Sitewide Infrastructure

Resource	Estimated Use	Capacity	Available Capacity
Electricity			
Power consumption (megawatt hours per year)	310,000	4,400,000	4,100,000
Peak load (megawatts)	60	500	440
Fuel[a]			
Oil (gallons per year)	410,000	N/A[b]	N/A
Potable Water (gallons per year)	320,000,000	2,950,000,000	2,630,000,000
Sewage (gallons per year)	250,000,000	383,000,000[c]	133,000,000

N/A – not applicable

[a] Oil use is for A-, D-, and K-Areas.

[b] Capacity is generally not limited, as delivery frequency can be increased to meet demand.

[c] Capacity includes the Central Sanitary Wastewater Treatment Facility and smaller treatment units in D-, K-, and L-Areas.

Note: To convert gallons to liters, multiply by 3.7854; miles to kilometers, multiply by 1.6093; tons (short) to metric tons, multiply by 0.90718. Totals are rounded to two significant figures.

Source: DOE 2015a

Electricity – Most of the electrical power consumed by SRS is generated by offsite coal-fired and nuclear power plants, and is supplied by the South Carolina Electric and Gas Company. Approximately 310,000 megawatt-hours per year of electricity is used at SRS, with an available capacity of 4,400,000 megawatt-hours per year (DOE 2015a). The peak load use is estimated to be 60 megawatts, with a peak load capacity of 500 megawatts.

Fuel – Biomass and a small amount of fuel oil are used at SRS to produce steam. The steam plant in A-Area, which burned coal, is no longer used and was replaced with a 30,000 pounds per hour (PPH) biomass plant in 2008. SRS also replaced its aging fuel-oil-fired package boilers in K- and L- Area with two small biomass 10,500 PPH heating plants in 2010 and the site's 1950's-vintage D-Area coal fire cogeneration plant with a 240,000 PPH biomass cogeneration facility (BCF) in 2012. The coal fired H-Area Powerhouse (built in early 1950s) was placed in "cold standby" condition in March 1995 but is no longer a viable source of steam. With the start-up of the BCF, SRS no longer uses coal as a source of fuel to produce steam (SRNS 2014b). Biomass is delivered by truck to SRS from a local 50 mile (80 kilometer) radius and the BCF has an approximate 30 day stockpile of biomass wood chips. All four of the biomass plants can burn fuel oil as a backup. An estimated 260,000 tons (236,000 metric tons) of biomass and 660,000 gallons (2.5 million liters) of fuel oil are burned at the four operating SRS biomass plants. Fuel oil is also used to power emergency generators. Fuel oil supplies can be delivered by truck or rail as needed. Furthermore, temporary storage tanks can be installed to supplement fuel consumption needs during construction activities. Thus, the capacity for biomass or fuel oil utilization is generally not considered to be limited. Natural gas is not used at SRS (DOE 2015a).

Water - The source of potable water at SRS is groundwater which is treated at facilities in A- and B-Areas and distributed to other areas of the site via a 27-mile (43-km) pipeline system. Annual water consumption (primarily process water of groundwater origin) is approximately 320 million gallons (1.78 billion liters), whereas the potable water production capacity at SRS is approximately 3.0 billion gallons (3.79 billion liters) (DOE 2015a).

Sewage - Sewage is treated at the Central Sanitary Wastewater Treatment Facility (CSWTF), located on Burma Road. This facility collects and treats 97 percent of sanitary wastewater generated at SRS. Approximately 18 miles (29 kilometers) of pressurized sewer line and 12 lift stations are used to transport sanitary waste to the CSWTF. The balance of the sanitary waste is treated at 3 smaller, independent facilities located in D-, K-, and L-Areas. Collectively, the sanitary systems include the CSWTF, the 3 smaller treatment facilities, 46 lift stations, and 58 miles (93 kilometers) of sewer pipe. The CSWTF and the smaller treatment units are estimated to collect and treat approximately 250 million gallons (950 million liters) of sewage per year with a capacity to treat up to 383 million gallons (1.5 billion liters) per year of sewage (DOE 2015a).

4 IMPACT ANALYSIS

Chapter 4 describes the environmental impacts of the alternatives evaluated in this Draft EA. Impacts from the Proposed Action are described in Section 4.1. Impacts on the Global Commons are described in Section 4.1.1, impacts at Joint Base Charleston-Weapons Station are described in Section 4.1.2, and impacts at the Savannah River Site (SRS) are described in Section 4.1.3. For SRS, those resource areas having the greatest potential for environmental impacts are discussed in Sections 4.1.3.1 through 4.1.3.6. These include; air quality, human health, socioeconomics, waste management, transportation, and environmental justice. Impacts on remaining resource areas are addressed in Section 4.1.3.7, Other Resources (including land use, visual resources, geology and soils, water resources, noise, ecological resources, cultural resources, and infrastructure). Impacts from the No Action Alternative are described in Section 4.2. Cumulative impacts are addressed in Section 4.3. Sections 4.4, 4.5, and 4.6 address irreversible and irretrievable commitments of resources, the relationship between local short-term uses of the environment and the maintenance and enhancement of long-term productivity, and mitigation, respectively.

Because technologies for processing the spent nuclear fuel from Germany are in various stages of development, DOE recognizes that there is uncertainty in their performance and therefore, potential impacts. In evaluating the potential environmental impacts of the processes, uncertainty is addressed in two ways. The first is to use conservative estimates of the parameters related to the processes (that is, use parameter values that tend to overestimate the potential environmental impacts). The second is to correlate the proposed processes to other similar and more completely characterized or previously evaluated processes.

The underlying chemistry for the proposed process technologies is well understood and serves as the basis for estimates of facility, equipment, and material requirements; processing rates; emissions; and waste generation. Additional information relevant to the analysis of impacts is derived from safety documents, site environmental reports, and previous NEPA analyses; relevant source documents are cited in the appropriate sections of this chapter. As an example, H-Canyon has a long history of dissolving spent nuclear fuel and recovering selected isotopes using the solvent extraction process, as well as down blending recovered isotopes to a specific concentration. Existing safety and NEPA documents addressing these well-established processes provide a solid foundation for evaluating the potential impacts of using these same or similar processes for the spent nuclear fuel from Germany. In the case of the L-Area melt and dilute process, the technology was previously studied and evaluated for another spent fuel type. Those studies and evaluations contribute to the understanding of the process steps, equipment requirements, and operating parameters. Understanding of existing L-Area operations and comparison to the evaluation for the other fuel provides a basis for estimates of the potential impacts of this technology.

Carbon digestion technologies are newly proposed and there are not existing processes or previous analyses that are directly comparable. Because the process chemistry is well understood, estimates of relevant process parameters (for example, emissions and waste generation) are believed to be reliable and conservative. DOE also recognizes that certain parameters, such as air emissions, can be controlled during the design phase. If during technology development, testing reveals higher emissions than those assumed for this analysis, additional control technologies could be added to the design of the air treatment system to ensure that emissions are reduced to levels that would comply with applicable standards and would also be as low as reasonably achievable. Based on engineering estimates intended to be conservative from an impacts perspective, facility operating experience, and comparison to previous analyses, DOE expects the actual impacts to be similar or less than those presented in this EA.

4.1 IMPACTS FROM THE PROPOSED ACTION

4.1.1 Impacts on the Global Commons

4.1.1.1 Impacts on the Global Commons under Incident-Free Transport

There would be no release of radioactive material under incident-free transport, meaning that there would be no radiological impacts on the global commons, including impacts on marine biota and fisheries from the proposed action. There would be minimal nonradiological impacts as discussed in this section.

Although there would be emissions of nonradiological air pollutants to the air from maritime vessels, the total number of shipments of spent nuclear fuel is not expected to exceed 30, with up to 8 shipments in a single year. For comparison, several thousand vessels annually traverse the global commons, and between 35 and 45 vessels are received annually at Joint Base Charleston – Weapons Station (Galen 2015). In 2011, 14,432 large ocean vessels made port calls in the South Atlantic Coastal Region (all ports from Alexandria, Virginia, to Miami, Florida) (DOT 2013a). During that year, there were 1,876 commercial vessel calls at the Port of Charleston (DOT 2013b) as well as 68 cruise ship departures (DOT 2013c). Given the small number of spent nuclear fuel shipments compared to the total number of vessels that annually traverse the global commons or call at the Port of Charleston, the shipments evaluated in this EA are not expected to appreciably add to global emissions of airborne pollutants.

For similar reasons, there would be minimal impacts from discharges of liquid effluents to ocean waters. Discharges, such as bilge water, from ships transporting spent nuclear fuel would be no larger than discharge from ships transporting other cargo, and there would be far fewer ships than the number of vessels that annually traverse the global commons or call at the Port of Charleston. Discharges in the Port of Charleston and the Cooper River (the location of Joint Base Charleston-Weapons Station), if any, would be restricted in accordance with applicable laws and requirements.

4.1.1.2 Human Health Impacts of Incident-Free Transport

The public would not receive a radiation dose from incident-free ocean transport of spent nuclear fuel; however, radiological impacts could be experienced by the crews of the ships carrying the spent fuel. The radiological impacts would depend on the duration of the voyages. As discussed in Chapter 2, Section 2.1.3.1, a 15-day voyage was assumed for a shipment from a German seaport.

This EA addresses the potential impacts from 30 shipments of spent nuclear fuel to Joint Base Charleston-Weapons Station occurring over approximately 3.5 years, with each shipment transporting 8 to 16 CASTOR casks secured within International Organization of Standardization (ISO) shipping containers. Some of a vessel's crew could be exposed to radiation while loading the containers of spent nuclear fuel onto the ship, while performing daily inspections of the vessel's cargo, and while unloading the shipping containers at Joint Base Charleston-Weapons Station. It is assumed that operational procedures for loading and unloading the shipping containers would be the same as those described in the *Final Environmental Impact Statement on a Proposed*

Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (*FRR SNF EIS)* for ocean shipment of FRR SNF (DOE 1996a),, and that the stowed shipping containers would be separated from each other in the cargo hold. The stowed cargo would be inspected by a 2-member crew on a 4-hour basis (i.e., 6 inspections per 24-hour period) (DOE 2014a), and each inspection of cargo consisting of 16 CASTOR casks would require 1 hour. A dose rate of 1 millirem per hour at 1 meter from any cask surface was conservatively assumed; no credit was taken for any shielding that could be provided by the impact limiters placed on both ends of the casks or by the ISO shipping containers. Finally, similar to the *FRR SNF EIS*, it was assumed that ship crew members loading, unloading, and inspecting the shipping containers would be exposed to radiation from the particular shipping container being handled as well as from other stowed shipping containers.

Table 4-1 presents estimated doses and risks, in terms of latent cancer fatalities (LCFs) that would be incurred by crew members during loading or unloading the shipping containers. **Table 4-2** presents estimated doses and risks that would be incurred by crew members during the daily cargo inspections; **Table 4-3** presents the estimated sum of doses and risks that would be incurred by all crew members. Similar to the *FRR SNF EIS* (DOE 1996a), it is assumed that five crew members would be exposed to ionizing radiation during loading and unloading operations (Chief Mate, Mate on Watch, Bosun, and two Seamen) and two crew members would be exposed during daily cargo inspections. Each table presents doses and risks assuming: (1) a single shipment of 8 or 16 casks, (2) 8 shipments of 8 or 16 casks in a single year, and (3) all shipments (455 casks) over approximately 3.5 years.

Radiological Impacts

In this EA, radiological consequences of operations and accidents are reported as doses and latent cancer fatalities (LCFs). An LCF is a death from cancer resulting from, and occurring some time after, exposure to ionizing radiation. A factor of 0.0006 LCFs per rem or person-rem is used to calculate the risk associated with radiation doses (DOE 2003); for acute individual doses above 20 rem, the risk factor is doubled (NCRP 1993).

For a group (for example, the offsite population), doses are reported in person-rem and LCFs are reported as a whole number, representing the number of people in the group statistically expected to develop an LCF as a result of the exposure. When the value calculated by multiplying the dose by the LCF risk factor of 0.0006 is less than 1, the reported value is rounded to 0 or 1 and the calculated value is shown in parentheses. For an individual, doses are reported in rem or millirem, along with the risk or likelihood of the dose resulting in an LCF. Because it is assumed that there is some level of risk associated with radiation exposure, regardless of the magnitude, the individual risk is not reported as 0.

Table 4-1: Doses and Risks to Ship Crew Members During Loading or Unloading Operations

Number of Casks	Dose [a] and LCF Risk to Individual Crew Members								Total Crew Dose[a] and LCFs	
	Chief Mate		Mate on Watch		Bosun		Seaman[b]			
	Dose (millirem)	LCF Risk	Dose (millirem)	LCF Risk	Dose (millirem)	LCF Risk	Dose (millirem)	LCF Risk	Dose (person-rem)	LCFs[c]
A Single Shipment										
8	1.7	1×10^{-6}	0.93	5×10^{-7}	1.7	1×10^{-6}	3.4	2×10^{-6}	0.011	0 (7×10^{-6})
16	3.4	2×10^{-6}	1.9	1×10^{-6}	3.4	2×10^{-6}	6.7	4×10^{-6}	0.022	0 (1×10^{-5})
8 Shipments in a Year										
64 (8 per shipment)	13	8×10^{-6}	7.5	4×10^{-6}	13	8×10^{-6}	27	2×10^{-5}	0.088	0 (5×10^{-5})
128 (16 per shipment)	27	2×10^{-5}	15	9×10^{-6}	27	2×10^{-5}	54	3×10^{-5}	0.18	0 (1×10^{-4})
All Shipments (Over approximately 3.5 Years)										
455	96	6×10^{-5}	53	3×10^{-5}	96	6×10^{-5}	190	1×10^{-4}	0.63	0 (4×10^{-4})

LCF = latent cancer fatality.

a Doses are determined assuming that the radiation levels of all CASTOR casks are 1 millirem per hour at 1 meter (3.3 feet) from the cask surfaces. Consistent with the *FRR SNF EIS* (DOE 1996a), crew members loading or unloading a shipping container were assumed to be exposed to radiation from the shipping container being handled as well as radiation from other shipping containers on the vessel.

b For each voyage, two seamen would receive radiation doses while loading cargo; the doses presented are per seaman.

c The reported values are the number of LCFs expected to occur in the ship crew population and are presented as whole numbers; the values in parentheses are the calculated values.

Note: Risks were determined using a factor of 0.0006 LCFs per rem or person-rem and are presented using one significant figure (DOE 2003).

As shown in Table 4-1, it is possible, although unlikely, that a crew member involved in loading or unloading operations could receive a radiation dose exceeding 100 millirem in a year, conservatively assuming the same crew members would be engaged in all loading and unloading operations. For example, a crew member (a seaman) hypothetically involved in all loading operations assuming 16 casks for each of 8 shipments in a single year would receive a total dose of 54 millirem; if this crew member were also involved with all unloading operations during this year, the total dose would be 110 millirem.

To mitigate potential radiation impacts to workers, NNSA would extend the program described in the mitigation action plan for FRR SNF to these shipments. Under the mitigation program applied to shipments of FRR SNF (DOE 1996c), NNSA requires that its shipping contractor obtain radiation surveys of FRR SNF casks before shipment, and use these data to ensure that the estimated dose to any crew member does not exceed 100 millirem in a year. NNSA also maintains a database of the actual radiation surveys for each cask and shipment, and includes clauses in its shipping contracts to minimize the likelihood that any member of a ship's crew would be exposed to more than 100 millirem during a single year

Cargo inspections would be performed six times daily (during each watch) while at sea, so the same individuals would not be involved in all daily cargo inspections. Therefore, the individual doses listed in **Table 4-2** could not be incurred by a single crew member but were assumed to be spread among six crew members.[27] As shown in Table 4-2, it is unlikely that any individual crew member involved in inspections would receive a radiation dose exceeding 100 millirem in a year, even if the same crew members participated in inspections and were aboard ship for all 8 shipments in a single year.

Table 4-2: Doses and Risks to Ship Crew Members During Daily Cargo Inspections

Number of Casks	Individual Crew Member		Inspection Crew	
	Dose (millirem)[a]	LCF Risk	Dose (person-rem)[a]	LCF[b]
A Single Shipment				
8	4.2	3×10^{-6}	0.025	0 (2×10^{-5})
16	9.5	6×10^{-6}	0.057	0 (3×10^{-5})
8 Shipments In a Year				
64 (8 per shipment)	34	2×10^{-5}	0.20	0 (1×10^{-4})
128 (16 per shipment)	76	5×10^{-5}	0.45	0 (3×10^{-4})
All Shipments (Over approximately 3.5 Years)				
455	270	2×10^{-4}	1.6	0 (1×10^{-3})

LCF = latent cancer fatality.

[a] Doses are determined assuming that the radiation levels at the surfaces of all casks are 1 millirem per hour at 1 meter (3.3 feet) from the cask surfaces. Crew members inspecting a shipping container are assumed to be exposed to radiation from the shipping container being inspected as well as radiation from other shipping containers that had been stowed.

[b] The reported values are the numbers of LCFs expected to occur in the inspection crew population and are reported as whole numbers; the values in parentheses are the statistically calculated values.

Note: Risks were determined using a factor of 0.0006 LCFs per rem or person-rem and are presented using one significant figure (DOE 2003).

As shown in **Table 4-3**, the total dose among all crew members could be up to 0.81 person-rem in a single year (assuming the maximum 8 shipments in a year with 16 casks per shipment). The total radiation dose among all crew members considering all shipments would be 2.9 person-rem. No LCFs would be expected (calculated value of 2×10^{-3}) as a result of this collective dose.

[27] Assuming that a member of a ship's crew works on a 4-hour-on, 8-hour-off basis, an individual crew member would perform cargo inspections twice daily. Therefore, 6 individual crew members could be involved in cargo inspections assuming each inspection involves 2 crew members and inspections are performed 3 times each 12-hour period. Persons performing cargo inspections are assumed to be ship officers and engineers (DOE 1996a, DOE 2014a).

Table 4-3: Doses and Risks to All Ship Crew Members

Number of Casks	Dose (person-rem)[a]	LCF[b]
1 Shipment		
8	0.047	0 (3×10^{-5})
16	0.10	0 (6×10^{-5})
8 Shipments		
64 (8 per shipment)	0.38	0 (2×10^{-4})
128 (16 per shipment)	0.81	0 (5×10^{-4})
All Shipments		
455	2.9	0 (2×10^{-3})

LCF = latent cancer fatality.

[a] Doses are the combination of doses among all crew members from loading cargo at the departure seaport, inspections during transit across the ocean, and unloading cargo at the destination seaport.

[b] The reported values are the numbers of LCFs expected to occur in the population of crew members and are reported as whole numbers; the values in parentheses are the statistically calculated values.

Note: Risks were determined using a factor of 0.0006 LCFs per person-rem and are presented using one significant figure (DOE 2003).

Shipping container handling and daily inspections would occur in accordance with radiation protection principles, and unauthorized crew members would have limited access to the radioactive cargo. Radiation doses received by crew members performing at-sea inspections of shipping containers could be reduced through careful spacing of the shipping containers, consistent with available stowage space. Additional shielding that might be provided by the proximity of other cargo cannot be predicted and is not considered in the exposure modeling conducted for this EA. Radiation doses associated with at-sea inspections could also be reduced by minimizing the amount of time taken for inspections consistent with the need to ensure cargo stability.

4.1.1.3 Human Health Impacts under Accident Conditions

Radiological Risks

Radiological risks to the global commons and crew members from an accident while at sea would be the product of (1) the probability of an accident of sufficient severity to cause the release of radioactive material from the casks, and (2) the consequences of the release of radioactive material. Because the fuel would be transported in very strong casks designed and certified to withstand routine transportation accidents with little or any release, only very rare, severe accidents would be expected to threaten the integrity of a cask and possibly result in a release of radioactive material. There would be nothing about the shipments of the spent nuclear fuel from Germany that would engender a greater probability of a severe accident than that associated with transporting other cargo. The potential radiological risks of a severe accident are summarized in this section.

Radiological Impacts on the Global Commons—In the unlikely event of a severe accident at sea, casks containing spent nuclear fuel could be released into the ocean and possibly ruptured. The response to and potential impacts of such an accident would depend on the location and condition of the packages following the accident (DOE 1994a, 2004). Casks that do not sink below about 200 meters (660 feet) could be located and recovered. Casks that are not damaged by the accident and sink deeper than about 200 meters (660 feet) could be breached by the pressure of the overlying water or, over time, by corrosion, and their contents released. The impacts from accidents at sea that involve a fire would be less than the impacts of the port accident discussed in Section 4.1.2.3.

The *FRR SNF EIS* (DOE 1996a) includes a detailed analysis of the potential impacts on the public and marine life from an at-sea accident involving a shipment of spent nuclear fuel. The potential impacts on the global commons from an accident during transport of the spent nuclear fuel from Germany to the United States were evaluated using techniques and assumptions similar to those in the *FRR SNF EIS*.

In the *FRR SNF EIS*, radiological impacts were evaluated for two high-consequence accident scenarios, vessels sinking in coastal and deep ocean waters with both a damaged and an undamaged cask that is not retrieved, and three types of fuel. The largest impacts were for an accident involving a Pegase cask loaded with Belgian Reactor (BR)-2 fuel containing 15.5 kilograms of heavy metal and 930,000 curies of radioactive material. A typical CASTOR cask with spent nuclear fuel from Germany is expected to contain about 4,500 curies[28] (SRNL 2014d). In the case of an accident at sea, the pathway of interest is ingestion. As a result and assuming all other factors remain the same, a comparison of ingestion dose conversion factors (associated with dispersal of radionuclides in seawater) and the radionuclide inventories indicates that human dose from ingestion of radionuclides released from a sunken CASTOR cask would be about a factor of 50 lower than a similar release from a Pegase cask containing BR-2 fuel. Although the simple analysis indicates that a cask of FRR BR-2 fuel used in the *FRR SNF EIS* impact analysis has about a 50 (47.8) times higher dose potential that an average cask of German fuel, the uncertainties in this approach require additional conservatisms. The differences in the overall characteristics of the spent fuel—German fuel is higher burnup but also has been out of the reactor long enough that the short-lived isotopes have decayed away while the BR-2 fuel was low burnup, but assumed to have only been out of the reactor 6 months—makes comparisons complicated. For analysis purposes, the dose assuming a release of radioactive material from a CASTOR cask was conservatively assumed to be a factor of 10 less than that from a cask of BR-2 fuel.

Table 4-4 summarizes the projected doses to individuals and marine life from accidents in coastal waters and the deep ocean that result in sunken casks. These values are the impacts for a single cask. For spent nuclear fuel from Germany, 16 casks per shipment are assumed and each cask is subject to leaking in one of these accidents. Analyses in the *FRR SNF EIS* assumed up to 2 casks per shipment. These projected dose rates are based on the corrosion rates for aluminum-clad fuel presented in the *FRR SNF EIS* (DOE 1996a); however, it is expected that long-term degradation rates for the spent nuclear fuel from Germany (graphite matrix) from exposure to sea water would be lower than the long-term corrosion rates for the aluminum-clad fuel used for the estimates in the *FRR SNF EIS*. Considering these and other conservative assumptions, actual dose rates to

[28] Although the Pegase casks with BR-2 fuel were projected to contain many more curies than the CASTOR casks, most of the activity in the Pegase casks is associated with short-lived isotopes. Most of the short-lived isotopes in the spent nuclear fuel from Germany have decayed.

individuals should be much lower than those projected in Table 4-4, and direct exposure dose to marine life even lower[29].

Table 4-4: Coastal and Deep Ocean Dose Rate Estimates for Accidents Resulting in Sinking of Undamaged and Damaged Casks

Dose Assuming the Accident Occurs (per cask) [a]	Accident and Sinking in Coastal Waters		Accident and Sinking in Deep Ocean	
	Cask of BR-2 Fuel[b]	German Fuel Cask[c]	Cask of BR-2 Fuel[b]	German Fuel Cask[c]
Undamaged Cask Peak Individual Dose (rem/yr)	0.19	0.019	cask fails[d]	cask fails[d]
Damaged Cask Peak Individual Dose (rem/yr)	14	<1.4	0.114	<0.0114
Undamaged Cask Peak Biota Dose (Fish) (rad/yr)	0.077	<0.0077	cask fails[d]	cask fails[d]
Damaged Cask Peak Biota Dose (Fish) (rad/yr)	0.62	<0.062	640	<64
Undamaged Cask Peak Biota Dose (Crustaceans) (rad/yr)	0.081	<0.0081	cask fails[d]	cask fails[d]
Damaged Cask Peak Biota Dose (Crustaceans) (rad/yr)	0.66	<0.066	880	<88
Undamaged Cask Peak Biota Dose (Mollusks) (rad/yr)	0.21	<0.021	cask fails[d]	cask fails[d]
Damaged Cask Peak Biota Dose (Mollusks) (rad/yr)	14	<1.4	30,000	<3000

BR-2 = Belgian Reactor-2; FRR = foreign research reactor; yr = year.

[a] In an accident, up to 2 FRR casks or 16 German fuel casks could sink. The total impacts could be proportionally higher if radionuclides were released from more than one cask.

[b] From Tables C-15 and C-16 of Appendix C of the *FRR SNF EIS* (DOE 1996a).

[c] Based on Tables C-15 and C-16 of Appendix C of the *FRR SNF EIS* (DOE 1996a); adjusted for the radionuclide inventories within CASTOR casks containing spent nuclear fuel from Germany (SRNL 2014d).

[d] Casks that are undamaged in the accident and sink deep in the ocean are assumed to fail and have the same impacts as the damaged casks.

The consequence estimates in Table 4-4 are indicative of what could happen if a spent nuclear fuel cask were to become submerged in coastal waters or in the deep ocean and not recovered. By combining an estimate of the frequency at which such a situation is expected to occur with the consequence estimates, an estimate of the risk associated with ocean transportation can be developed. The accident probabilities and assumptions used in the *FRR SNF EIS* (DOE 1996a) were adapted to the proposed shipment of 455 CASTOR casks to estimate the probabilities of accidents that could result in a CASTOR cask sinking in coastal waters or the deep ocean. The probabilities of accidents during ocean transport were based on accident frequencies used in the *FRR SNF EIS*. The probability of an accident that could result in a CASTOR cask being submerged in coastal waters was estimated to be 2.9×10^{-11} for a damaged cask, and 1.5×10^{-8} for an undamaged cask. The probability of an accident that could result in a CASTOR cask being submerged in deep ocean waters was estimated to be 1.1×10^{-6} (the cask was assumed to be damaged). Using these accident probabilities and the estimated annual doses assuming an accident

[29] As indicated in the *FRR SNF EIS*, the estimated dose rates are very conservative (DOE 1996a: Section C.5.4). The radioactive material was assumed to be quickly released to the open water once casks became corroded, and no credit was taken for the possibility that casks would likely become buried in silt. Additionally, once released from the casks, the radioactive material was assumed to be transported over short distances. This assumption results in high estimated doses to organisms, especially mollusks, in the vicinities of the casks.

occurred, radiological risks were calculated as *dose-risks* to humans and marine life, which are determined as the products of the probability of an accident times the annual doses assuming the accident occurred.

Table 4-5 presents the dose-risk estimates to individuals and marine life for at-sea accidents in coastal waters or the deep ocean resulting from a sunken cask. The overall accident risks in the global commons from ship accidents associated with the transport of spent nuclear fuel from Germany to the United States are about 15 times lower than those projected for the *FRR SNF EIS*. This is due to the lower inventories of radionuclides in the CASTOR casks (with doses from the CASTOR casks estimated to be a factor of 10 lower than the FRR casks) and the fewer number of casks shipped (455 CASTOR casks versus 721 casks with all types of FRR fuel). An accident at sea that caused sufficient damage to the casks to release some of the radioactive content could cause radiological impacts on crew members. These impacts would be highly specific to the accident scenario and the locations and actions of affected crew members. If the accident involved a collision with another ship, it is hypothesized that the collision could cause a breach and/or severe fire. The probability of a collision between ships is less at sea than in congested areas such as ports, channels, and rivers. This postulated accident would cause immediate nonradiological risk and also threaten the seaworthiness of the vessel. Either situation would put the crew at more immediate risk to life than would release of radioactive material.

Table 4-5: Radiological Dose-Risk Estimates for At-Sea Accidents

Dose-Risk	Accident and Sinking in Coastal Waters		Accident and Sinking in Deep Ocean	
	FRR (BR-2) Fuel[a]	German Fuel[b]	FRR (BR-2) Fuel[a]	German Fuel[b]
Undamaged Cask Peak Individual Dose-Risk (millirem/yr)	4.4×10^{-6}	2.8×10^{-7}	cask fails[c]	cask fails[c]
Damaged Cask Peak Individual Dose-Risk (millirem/yr)	6.4×10^{-7}	4.1×10^{-8}	1.9×10^{-4}	1.2×10^{-5}
Undamaged Cask Peak Biota Dose-Risk (Fish) (millirad/yr)	1.8×10^{-6}	1.1×10^{-7}	cask fails[c]	cask fails[c]
Damaged Cask Peak Biota Dose-Risk (Fish) (millirad/yr)	2.9×10^{-8}	1.8×10^{-9}	1.1	6.7×10^{-2}
Undamaged Cask Peak Biota Dose-Risk (Crustaceans) (millirad/yr)	1.9×10^{-6}	1.2×10^{-7}	cask fails[c]	cask fails[c]
Damaged Cask Peak Biota Dose-Risk (Crustaceans) (millirad/yr)	3.0×10^{-8}	1.9×10^{-9}	1.5	9.2×10^{-2}
Undamaged Cask Peak Biota Dose-Risk (Mollusks) (millirad/yr)	4.8×10^{-6}	3.1×10^{-7}	cask fails[c]	cask fails[c]
Damaged Cask Peak Biota Dose-Risk (Mollusks) (millirad/yr)	6.4×10^{-7}	4.1×10^{-8}	51	3.1

BR-2 = Belgian Reactor-2; FRR = foreign research reactor.

a From Tables C-17 and C-18 of Appendix C of the *FRR SNF EIS* (DOE 1996a)

b Based on Tables C-17 and C-18 of Appendix C of the *FRR SNF EIS* (DOE 1996a); adjusted for the radionuclide inventories within CASTOR casks containing spent nuclear fuel from Germany (SRNL 2014d).

c Casks that are undamaged in the accident and sink deep in the ocean are assumed to fail and have the same impacts as the damaged casks.

Nonradiological Risks

Nonradiological Impacts on the Global Commons. It is possible that a ship containing spent nuclear fuel could pass through an area routinely inhabited by the northern right whale (*Eubalaena glacialis*), a federally endangered species that is protected internationally. Compliance with the International Maritime Organization, Coast Guard, and National Marine Fisheries Service speed and reporting requirements described in Section 3.1, Global Commons, would mitigate impacts from the proposed shipments. Another possibility is a strike by a ship carrying spent nuclear fuel on an endangered species such as a sea turtle or manatee (*Trichechus manatus*); both species are found in the vicinity of Charleston, South Carolina. The potential for ship strikes can be reduced by adherence to speed restrictions in port entrance channels and port reaches.

Nonradiological Impacts on Ship Crew Members. Shipments of spent nuclear fuel from Germany to the United States would not present meaningful nonradiological risks to ship crews. There would be nothing inherent in shipping the spent nuclear fuel from Germany that would involve more risk than would be involved in transporting other cargo. The only nonradiological risk that could arise from shipping spent nuclear fuel would result from the hypothetical shifting of cargo within the vessel to the point of injuring crew members or jeopardizing the seaworthiness of the vessel. This risk, however, would be independent of the spent nuclear fuel. There would be nothing about the physical characteristics of the spent nuclear fuel from Germany that would present additional difficulties in safely securing the shipping containers for marine transport.

4.1.1.4 Intentional Destructive Acts on the Global Commons

Maritime areas where acts of terrorism or piracy are more likely would be avoided or ships passing thorough these areas would invoke additional security measures as necessary. About 80 percent of all acts of piracy take place in the territorial waters of sovereign nations. In 2007, the locations having the most incidents of piracy included waters near Indonesia, Nigeria, and Somalia (Petretto 2008). Shipments of spent nuclear fuel from Germany to the United States would not transit waters near these nations.

If an intentional destructive act were to occur at sea, potential impacts would primarily be to onboard personnel. Potential impacts could range from fatalities associated with explosions or drowning to lesser impacts of radiation exposure to untrained or uninformed personnel in the immediate vicinity of the shipping containers containing spent nuclear fuel. If the intentional destructive act occurred near a coastline and caused the release of radioactive material into the air or water, radiological impacts on people on land would be less than those of a severe accident at Joint Base Charleston-Weapons Station (see Section 4.1.2.3).

4.1.2 Impacts on Joint Base Charleston – Weapons Station

4.1.2.1 Nonradiological Impacts from Incident-Free Seaport Operations

Shipments of spent nuclear fuel from Germany to the United States would not noticeably affect the volume of ship traffic into or out of the Charleston area, meaning that the shipments would have little effect on resource areas such as water quality, marine life, or socioeconomics. Up to 8 shipments of spent nuclear fuel are expected in a single year. These shipments would represent less than 1 percent of the 1,944 large commercial vessel and cruise ship calls at the Port of Charleston in 2011 (DOT 2013b, 2013c). [30]

At Joint Base Charleston-Weapons Station, existing infrastructure would be used to manage the shipments of spent nuclear fuel, with no need for construction or modification of seaport facilities, and with no land disturbance that could potentially affect land use, biological resources, cultural resources, or geologic media. The same types and quantities of nonradioactive wastes and pollutants, including greenhouse gases discharged to the air, would be generated as those associated with normal operation of ships and port facilities. Given the small number of annual shipments, there would be no meaningful additional use of utilities such as water or electricity beyond those currently needed for port operation. Any discharges to surface water arising from port operations would be expected to be in compliance with permitted levels. Because work would be accomplished using existing DOE, seaport, and contractor personnel, shipments of spent nuclear fuel from Germany would not affect socioeconomic conditions in the Charleston area.

4.1.2.2 Radiological Impacts from Incident-Free Seaport Operations

Under incident-free transport conditions, there would be no release of radioactive material to air or water, and no generation of radioactive waste. Because Joint Base Charleston-Weapons Station is a secure site where unauthorized personnel would be excluded from areas where the containers would be transferred from ships to rail cars, members of the public would not be in proximity to the cargo and would not receive any radiation dose. Because members of the public would be protected from radiological risk, no disproportionately high and adverse radiological risks would occur among low-income and minority populations in the vicinity of Joint Base Charleston-Weapons Station.

Radiation doses at the seaport could be received by workers other than ship crews (i.e., workers removing the shipping containers from the vessels and transferring them to rail cars for transport to SRS).[31] Doses and risks from shipping the spent nuclear fuel from Germany are presented in **Table 4-6**.[32] No worker is expected to receive a dose exceeding 100 millirem in a year. The total dose among all workers[33] is projected to be approximately 0.24 person-rem, with no LCFs expected from this dose (calculated value of 1×10^{-4} LCF).

Although the radiation dose to dock workers is expected to be low as shown in Table 4-6, to maintain worker doses within applicable standards and reduced to levels as low as reasonably achievable (ALARA), DOE would adopt the same radiation protection procedures for the receipt

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[30] To reach Joint Base Charleston – Weapons Station, ships must travel up the Cooper River past the port of Charleston. The number of annual military vessel calls at Joint Base Charleston – Weapons is classified.

[31] Ship crew members are assumed to assist in removal of the shipping containers from the vessels; the doses and risks received by crew members from vessel unloading activities are included with the doses and risks evaluated in Section 4.1.1.2.

[32] Estimated doses received by cargo handlers and staging personnel are consistent with the assumption in the *FRR SNF EIS* that unloading activities would require 65 minutes per shipping container (DOE 1996a). Experience with the FRR SNF Acceptance Program suggests that the actual unloading time would be closer to 20 minutes per shipping container (DOE 2009a). The less time required to unload the shipping containers, the smaller the radiation dose received by cargo handlers and other involved personnel.

[33] Consistent with the *FRR SNF EIS* analysis (DOE 1996a), the number of seaport workers receiving radiation doses from cargo unloading and transfer operations is assumed to be 14; however, only 10 to 12 seaport workers may actually participate in these operations considering experience under the FRR SNF Acceptance Program (DOE 2014a).

and transfer of the spent nuclear fuel from Germany that are routinely employed under the FRR SNF Acceptance Program. Personnel involved in unloading and package transfer operations at the seaports would be monitored by radiation safety technicians who would ensure compliance with applicable requirements (DOE 2009a).

Table 4-6: Incident-Free Impacts for Unloading Shipping Containers of Spent Nuclear Fuel from Chartered Ships

Risk Group[a, b, c]	Involved Worker		Worker Population	
	Dose (millirem)	LCF Risk[d]	Dose (person-rem)	LCFs[d]
1 Shipment – 8 to 16 Shipping Containers[e]				
Inspectors (6)	0.97	6×10^{-7}	0.0040	0 (2×10^{-6})
Port Cargo Handlers (4)	0.34	2×10^{-7}	0.0011	0 (7×10^{-7})
Port Staging Personnel (5)	0.21	1×10^{-7}	0.0034	0 (2×10^{-4})
Maximum	0.97	6×10^{-7}	NA	NA
Total	NA	NA	0.0085	0 (5×10^{-6})
8 Shipments – 8 to 16 Shipping Containers per Shipment[e]				
Inspectors (6)	7.8	5×10^{-6}	0.032	0 (2×10^{-5})
Port Cargo Handlers (4)	2.8	2×10^{-6}	0.0090	0 (5×10^{-6})
Port Staging Personnel (5)	2.4	1×10^{-6}	0.028	0 (2×10^{-5})
Maximum	7.8	5×10^{-6}	NA	NA
Total	NA	NA	0.068	0 (4×10^{-5})
All Shipments[e]				
Inspectors (6)	28	2×10^{-5}	0.11	0 (7×10^{-5})
Port Cargo Handlers (4)	9.8	6×10^{-6}	0.032	0 (2×10^{-5})
Port Staging Personnel (5)	8.5	5×10^{-6}	0.098	0 (6×10^{-5})
Maximum	28	2×10^{-5}	NA	NA
Total	NA	NA	0.24	0 (1×10^{-4})

LCF = latent cancer fatality; NA = not applicable.

[a] CASTOR cask dose rates were assumed to be 1 millirem per hour at 1 meter [3.3 feet] from the cask surface.

[b] Results are based on the conservative assumption that port personnel handling a shipping container would receive radiation exposures from that shipping container as well as radiation exposures from other shipping containers on the vessel.

[c] Numbers in parentheses are the assumed numbers of exposed personnel in each risk group.

[d] The reported values are the number of LCFs expected to occur in the worker population and are presented as whole numbers; the values in parentheses are the statistically calculated values.

[e] Reported values are for a shipment of 16 casks. For a shipment of 8 casks, impacts would be half of those shown.

Source: DOE 1996a (with dose rate adjusted to 1 millirem/hour at 1 meter [3.3 feet] from the container surface).

Note: Totals may not equal the sums of table entries due to rounding.

Risks were determined using a factor of 0.0006 LCFs per rem or person-rem and are presented using one significant figure (DOE 2003).

4.1.2.3 Human Health Impacts under Accident Conditions

Accidents associated with potential port activities at a range of U.S. ports were discussed and evaluated in Section 4.2.2 and Appendix D of the *FRR SNF EIS* (DOE 1996a). As considered in the *FRR SNF EIS*, the overall probability of a ship collision and cask breach (per shipment risk) depends on the number of voyages and stowed casks. In the *FRR SNF EIS*, a maximum of two casks in a single hold were assumed, but the potential risk from accidents was modeled assuming one cask per shipment. As stated in the *FRR SNF EIS*, the impacts of an accident with two casks in a hold could be twice as severe as the consequences of an accident involving one cask, but the per-voyage probability of an accident involving a ship carrying two casks would be half that for two ships each carrying a single cask.

More recent analysis has shown that the cask damage scenarios postulated in the *FRR SNF EIS* are very conservative (DOE 1998). In reality, a spent nuclear fuel cask is much stronger than the hull of the vessel carrying it. If there were a collision involving penetration of the hull of a vessel transporting spent nuclear fuel, a fuel cask would likely be pushed aside or out the other side of the vessel before enough force could be brought to bear on the cask to breach it. Although it is likely that a number of casks would survive a port accident undamaged, for this EA, impacts are reported on a per-cask basis regardless of the number of casks in a shipment.

In the *FRR SNF EIS*, radiological impacts were evaluated for port accident scenarios assuming three types of nuclear fuel. Similar to the analysis summarized in Section 4.1.1.3 of this EA, the largest impacts resulted from an accident involving a Pegase cask loaded with BR-2 fuel containing 15.5 kilograms of heavy metal and 930,000 curies of mostly short-lived radioactive material (DOE 1996a), whereas a typical CASTOR cask with spent nuclear fuel from Germany is expected to contain about 4,500 curies. For a port accident, the pathway of interest is the inhalation pathway. As a result the, assuming all other factors would remain the same, comparison of the inhalation dose conversion factors and the radionuclide inventories of a cask of BR-2 fuel and CASTOR casks indicates that the human dose from inhalation of airborne radioactive material released from a ship collision and fire and a port-area accident involving a CASTOR cask would be about a factor of 25 lower than that for a cask of BR-2 fuel. The differences in the overall characteristics of the spent fuel—German fuel is higher burnup but also been out of the reactor long enough that the short-lived isotopes have decayed away while the BR-2 fuel was low burnup but assumed to have only been out of the reactor 6 months—makes comparisons complicated. For analysis purposes, the dose per CASTOR cask was conservatively assumed to be a factor of 10 less than that for a cask of BR-2 fuel.

The principal analysis factor that changed since the *FRR SNF EIS* was issued is the population in the port area. For the analysis in this EA, the population in the port area was scaled to the year 2020 to reflect population increases since the 1990 census data used for the *FRR SNF EIS* (DOE 1996a).

Table 4-7 summarizes projected radiological impacts on individuals and the general population from a ship accident resulting in a severe fire that threaten the integrity of a Pegase cask containing BR-2 fuel and a CASTOR cask containing German fuel. Similar to the analysis in the *FRR SNF EIS* (DOE 1996a), accident impacts were determined for a maximally exposed individual assumed to be in the centerline of a plume and located at a distance of up to 1 mile (1.6 kilometers) from the assumed accident; population doses were determined for the population with a 50-mile (80-kilometer) radius of the assumed accident. The estimated impacts reflect the dose calculations and modeling assumptions used in the *FRR SNF EIS*, adjusted for the German fuel isotopic concentrations and CASTOR cask loading and population growth in the Charleston, South Carolina, area.

Impacts presented in Table 4-7 vary according to the accident release category and fuel type. Releases from the German fuel are estimated to result in doses to both the maximally exposed individual and the general population that are a factor of 10 lower than those for the comparable scenario for BR-2 fuel. The analysis does not take into account the differences in cask type or the characteristics of spent nuclear fuel from Germany that would provide an ability to survive a severe fire. The graphite surrounding the spent fuel kernels would be expected to provide more high-temperature protection for the fuel kernels and fission products in very severe fires than the aluminum-based fuels evaluated in the *FRR SNF EIS*, and have lower release rates. Therefore, the projected impacts in Table 4-7 are conservative in regard to the impacts of port fires involving spent nuclear fuel from Germany.

Table 4-7: Radiological Impacts from a Single Cask in the Event of a Port Accident

Accident	Accident Probability (per shipment)	Maximum Individual Dose[a] (rem)	Maximum Individual Probability of an LCF	Population Dose[a, b] (person- rem)	Population LCF[c]
FRR (BR-2) Fuel					
Charleston Port Fire, Release Category 4	6.5×10^{-6}	8.6×10^{-5}	5×10^{-8}	4.9×10^{-2}	$0\ (3\times 10^{-5})$
Charleston Port Fire, Release Category 5	5.0×10^{-9}	6.8×10^{-2}	4×10^{-5}	540	$0\ (3 \times 10^{-1})$
Charleston Port Fire, Release Category 6	6.0×10^{-10}	7.1×10^{-2}	4×10^{-5}	550	$0\ (3 \times 10^{-1})$
Charleston Population Dose-Risk per shipment[d]	N/A	N/A	N/A	3.4×10^{-6}	$0\ (2 \times 10^{-9})$
German Fuel[b]					
Charleston Port Fire, Release Category 4	6.5×10^{-6}	8.6×10^{-6}	5×10^{-9}	4.9×10^{-3}	$0\ (3 \times 10^{-6})$
Charleston Port Fire, Release Category 5	5.0×10^{-9}	6.8×10^{-3}	4×10^{-6}	54	$0\ (3 \times 10^{-2})$
Charleston Port Fire, Release Category 6	6.0×10^{-10}	7.1×10^{-3}	4×10^{-6}	55	$0\ (3 \times 10^{-2})$
Charleston Population Dose-Risk per shipment[d]	NA	NA	NA	3.4×10^{-7}	$0\ (2 \times 10^{-10})$

BR-2 = Belgian Reactor-2; FRR = foreign research reactor; LCF = latent cancer fatality; NA = not applicable.

[a] Updated from the *FRR SNF EIS* (DOE 1996a: Table D-31). Spent nuclear fuel from Germany impacts scaled from BR-2 fuel results to reflect different isotopic inventories. Sixteen casks per shipment could be subject to damage in a collision, but severe damage to multiple casks resulting in the reported per-cask impacts would not be likely (DOE 1998).

[b] Based on a Charleston, SC port-area population projected growth of 30 percent from 1990 to 2020.

[c] Numbers of LCFs in the population are whole numbers; the statistically calculated values are provided in parentheses when the reported result is less than 1.

[d] Determined as the sum over all accident categories of the accident category frequency times the dose per category.

Table 4-8 presents the population radiological risk estimates for port accidents for both the highest consequences BR-2 fuel evaluated in the *FRR SNF EIS* and the spent nuclear fuel from Germany. These risk estimates were compiled using the population doses presented in Table 4-7 and the accident probabilities per shipment for each severe accident release category. The population risk associated with shipping all 455 casks of spent nuclear fuel from Germany would be a factor of about 10 lower than the risks associated with 473 casks of BR-2 fuel as evaluated in the *FRR SNF EIS*. The lower risk is primarily due to the lower population doses estimates for the CASTOR casks compared to the FRR SNF casks.

The potential population exposures from port accidents are low enough to assure that any effect on plants and animals would be minimal (see DOE 1996a). As discussed in the *FRR SNF EIS*, if a cask or casks were sunk in coastal waters, DOE would locate and recover the cask(s), thus minimizing the potential impacts on marine life.

Table 4-8: Radiological Risk Estimates for Port Accidents

Shipment	Per Shipment Dose and Risk		Total Shipment Dose and Risk[a]	
	Population Dose (person-rem)	Population Risk (LCF)	Population Dose (person-rem)	Population Risk (LCF)
BR-2 Fuel[b]	3.4×10^{-6}	0 (2.0×10^{-9})	1.6×10^{-3}	0 (9.5×10^{-7})
German Fuel[c]	3.4×10^{-7}	0 (2.0×10^{-10})	1.5×10^{-4}	0 (9.8×10^{-8})

BR-2 = Belgian Reactor-2; LCF = latent cancer fatality.

[a] Assuming shipment of 473 casks of BR-2 fuel and 455 casks of German fuel.

[b] Updated from the *FRR SNF EIS* (DOE 1996a: Table D-31) evaluation of 473 shipments of BR-2 fuel, and a port-area population projected to 2020.

[c] Evaluation of 455 casks of spent nuclear fuel from Germany scaled from BR-2 fuel results to reflect different isotopic inventories and a port-area population projected to 2020.

4.1.2.4 Intentional Destructive Acts

It is not possible to predict the occurrence of sabotage or terrorism events or the exact nature of such events if they were to occur. Nonetheless, the *FRR SNF EIS* (DOE 1996a) examined three scenarios involving FRR SNF that if applied to spent nuclear fuel from Germany could have comparable impacts. Two scenarios involve explosive damage to shipping casks and one involves theft of a shipping cask. None of these scenarios would lead to a criticality accident because the contents of the casks are configured to avoid criticality. However, these scenarios could result in localized contamination.

Explosive Damage to a Shipping Cask—In one scenario, it was assumed that blast damage to a cask containing highly irradiated SNF would spread fuel elements on the ground, producing the highest possible direct dose rate. Based on this hypothetical, conservative analysis, an evacuation distance of about 900 meters (3,000 feet) was determined to be sufficient to maintain a dose rate of less than 10 millirem per hour (DOE 1996a).

In a second scenario, it was assumed that explosive penetration of a cask would cause damage of spent nuclear fuel inside the cask, with release of all noble gases and one percent of the solid spent nuclear fuel as airborne aerosols. Using the Melcor Accident Consequence Code System (MACCS) computer code, the impacts of this event were determined for the most populous port considered in the *FRR SNF EIS*, Elizabeth, New Jersey, with an 80-kilometer (50-mile) radius population of 16 million people. A population dose of 208,000 person-rem was estimated with no acute fatalities or short-term adverse health effects. Up to 91 LCFs were projected among the population, with an average individual lifetime radiation dose of about 200 millirem among the one to two million people who would be exposed (because this is an acute event, it was assumed that atmospheric conditions would cause impacts in mostly one direction, affecting people within a 45-degree angle sector) (DOE 1996a).

In 2009, the scenario was adjusted to reflect the conditions for Joint Base Charleston – Weapons Station. The 80-kilometer (50-mile) population around Joint Base Charleston – Weapons Station was projected to be approximately 1 million people as of 2020. The same cask radionuclide inventory was assumed as that in the *FRR SNF EIS*. The population dose for this revised scenario was 26,000 person-rem. Applying the current risk factor of 0.0006 LCFs per person-rem (DOE 2003), approximately 16 LCFs could be expected. The explosion itself would likely produce fatalities, injuries and property damage associated with blast impacts in the immediate vicinity of the cask (DOE 2009c).

Theft of a Shipping Cask—The *FRR SNF EIS* considered the scenario of theft of a spent nuclear fuel cask, although this occurrence is considered to be very unlikely due to the security measures that would be in place. In addition, the large size and weight (20 to 30 metric tons) of the cask and the inherent radioactivity of the spent nuclear fuel would deter most thefts. The cask could not be opened without great personal risk due to large radiation exposures. As discussed in the *FRR SNF EIS*, thieves would not be able to alter the fuel configuration inside the cask or have enough time or resources to change the moderating material to achieve criticality. If thieves were to remove the unshielded spent nuclear fuel, the resulting impacts on the public would be the same or less severe than other intentional destructive acts such as explosive damage to shipping casks.

4.1.3 Impacts on the Savannah River Site

4.1.3.1 Air Quality

Nonradioactive air pollutant impacts at SRS under each alternative are evaluated in this section. Radioactive air pollutant impacts at SRS are evaluated in Section 4.1.3.2.

Activities under the H- and L-Area Alternatives could result in emissions of criteria, hazardous, and toxic air pollutants from facility construction, operations, and employee travel. In order to evaluate the impacts of air emissions on the Air Quality Control Region, the emissions associated with the project activities were compared with the total emissions on a pollutant-by-pollutant basis using 2011 National Emissions Inventory (NEI) data. To provide a more conservative analysis, Aiken County was selected as the Air Quality Control Region instead of the EPA-designated Air Quality Control Region, which is a much larger area.

EPA's regulations for "Determining Conformity of General Federal Actions to State or Federal Implementation Plans" (EPA 2010; 40 CFR 93.150 – 93.165) require a conformity determination for certain-sized projects in nonattainment areas. A conformity determination is not necessary to meet the requirements of the conformity rule for the alternatives considered in this EA because SRS is located in an area that is in attainment for all criteria pollutants (EPA 2014c).

4.1.3.1.1 <u>H-Area Alternative</u>

Construction—Under the H-Area Alternative, approximately 0.4 acres of previously disturbed land would be disturbed for construction of new storage pads and roadways to store the CASTOR casks. Construction of the storage pads would not be expected to exceed existing permit levels for SRS portable heavy equipment operation (DOE 2014a).

No land disturbance or construction external to H-Canyon would be required for the Vitrification Option. Construction of the LEU Waste and LEU/Thorium Waste Options would include 5 acres of land disturbance and fugitive air emissions during construction of the uranium solidification facility. Heavy equipment would be operated during the 2-year construction period (DOE 2014a). **Table 4-9** shows the estimated annual construction emissions for the H-Area Alternative, LEU

Waste and LEU/Thorium Waste Options. Construction emissions would be less than 1 percent of Aiken County emissions.

Table 4-9: Estimated Annual Construction Emissions under the H-Area Alternative, LEU Waste and LEU/Thorium Waste Options

Source	Pollutant Emissions (tons/year)				
	VOC	SO_2	NO_2	CO	TSP
Diesel Equipment[a]	4.3	27	58	22	4.3
Construction Fugitive Emissions[a]	0.002	NA	NA	NA	24
Concrete Batch Plant[a]	NA	NA	NA	NA	2.9
TOTAL	4.3	27	58	22	31
2011 NEI Aiken County[b]	40,128	5,020	7,646	49,790	23,730
Percentage of Aiken County Emissions	0.01	0.54	0.76	0.04	0.13

CO = carbon monoxide; NA = not applicable; NEI = National Emissions Inventory; NO_2 = nitrogen dioxide; SO_2 = sulfur dioxide; VOC = volatile organic compounds; TSP = total suspended particulates

[a] Source: DOE 2014a

[b] Source: EPA 2014d

Operations—Under the H-Area Alternative, no changes in activities above normal maintenance activities and within the limits of permits for existing SRS portable heavy equipment operation are expected to result from the receipt, storage, and transfer of CASTOR casks. Therefore, no increase in air emissions is expected from receipt, storage, and transfer of CASTOR casks (DOE 2014a).

Table 4-10 shows estimated criteria air pollutant emissions under the Vitrification Option. The highest total emissions would be from nitrogen dioxide, and would represent 2.6 percent of Aiken County emissions for that pollutant. Although emissions are expected to be similar to historical levels and within current permitted levels (DOE 2014a), the change in nitrogen dioxide emissions would necessitate a permit review to determine whether revisions to the Title V Air Operating Permit (DOE 2007) would be required.

Table 4-10: Estimated Annual Criteria Air Pollutant Emissions under the H-Area Alternative, Vitrification Option

Source	Criteria Pollutant Emissions (tons/year)						
	VOC	SO_2	NO_2	CO	$PM_{2.5}$	PM_{10}	TSP
H-Canyon – Carbon Digestion[a]	NA	NA	30	206	NA	NA	NA
H-Canyon – Kernel Dissolution[a]	NA	NA	147	NA	0.32	0.32	0.32
Defense Waste Processing Facility[b, c]	0.00	0.00	0.09	0.01	NA	NA	0.00
Salt Waste Processing Facility[c, d]	56	0.26	17	4.3	0.00	0.32	0.76
Saltstone Production Facility[c, e]	NA	NA	NA	NA	0.03	0.03	0.03
TOTAL[f]	56	0.26	194	210	0.35	0.69	1.1
2011 NEI Aiken County[g]	40,128	5,020	7,646	49,790	7,217	23,730	23,730
Percentage of Aiken County Emissions	0.14	0.01	2.6	0.42	<0.01	<0.01	<0.01

CO = carbon monoxide; NA = not applicable; NEI =National Emissions Inventory; NO_2 = nitrogen dioxide; PM_{10} and $PM_{2.5}$ = particulate matter with a diameter of less than or equal to 10 microns and 2.5 microns, respectively; SO_2 = sulfur dioxide; VOC = volatile organic compounds, TSP=total suspended particulate

a Source: DOE 2014a.

b Source: DOE 1994b; adjusted for 100 days of operation.

c The listed operational durations represent the times required to process the spent nuclear fuel from Germany wastes pursuant to each major activity, which may require less than 1 year for some activities. For example, the time required for vitrification of HLW represents only the time required for vitrification of the HLW generated from that alternative option (about 100 days), and not the time required to process all SRS HLW at DWPF

d Source: DOE 2001a (Salt Waste Processing SEIS); adjusted for 24 days of operation.

e Source: DOE 2007; adjusted for 24 days of operation.

r This estimate is conservative because these activities may not occur during the same year.

g Source: EPA 2014d, (total countywide PM-10 emissions compared to TSP)

Table 4-11 shows estimated air emissions under the LEU Waste or LEU/Thorium Waste Options, which would be the same for either option. The highest total emissions would be from nitrogen dioxide, and would represent 2.6 percent of Aiken County emissions for that pollutant. Emissions from the LEU Waste or LEU/Thorium Waste Options are slightly higher than those for the Vitrification Option. The increase in nitrogen dioxide emissions would necessitate a permit review to determine whether revisions to the Title V Air Operating Permit would be required (DOE 2007).

Table 4-11: Estimated Annual Criteria Air Pollutant Emissions under the H-Area Alternative, LEU Waste or LEU/Thorium Waste Options

Source	Criteria Pollutant Emissions (tons/year)						
	VOC	SO_2	NO_2	CO	$PM_{2.5}$	PM_{10}	TSP
H-Canyon – Carbon Digestion[a]	NA	NA	30	206	NA	NA	NA
H-Canyon – Kernel Dissolution[a]	NA	NA	147	NA	NA	0.32	0.32
Solidification Facility (uncontrolled) [a]	0.01	NA	NA	NA	0.11	0.29	0.30
Defense Waste Processing Facility[b,c]	0.00	0.00	0.03	0.00	NA	NA	0.00
Salt Waste Processing Facility[c,d]	70	0.33	21	5.4	NA	0.40	0.95
Saltstone Production Facility[c, e]	NA	NA	NA	NA	NA	NA	0.04
TOTAL[f]	70	0.33	198	211	0.10	0.83	1.6
2011 NEI Aiken County[g]	40,128	5,020	7,646	49,790	7,217	23,730	23,730
Percentage of Aiken County Emissions	0.18	0.01	2.6	0.42	0.01	<0.01	0.01

CO = carbon monoxide; NA = not applicable; NEI =National Emissions Inventory; NO_2 = nitrogen dioxide; PM_{10} and $PM_{2.5}$ = particulate matter with a diameter of less than or equal to 10 microns and 2.5 microns, respectively; SO_2 = sulfur dioxide; VOC = volatile organic compounds, TSP=total suspended particulate

a Source: DOE 2014a.

b Source: DOE 1994b; adjusted for 30 days of operation.

c The listed operational durations represent the times required to process the spent nuclear fuel from Germany wastes pursuant to each major activity, which may require less than 1 year for some activities. For example, the time required for vitrification of HLW at DWPF represents only the time required for vitrification of the HLW generated from that alternative option (about 30 days), and not the time required to process all SRS HLW at DWPF.

d Source: DOE 2001a (Salt Waste Processing SEIS); adjusted for 30 days of operation.

e Source: DOE 2007; adjusted for 30 days of operation.

f\ This estimate is conservative because these activities may not occur during the same year.

g Source: EPA 2014d, (total countywide PM-10 emissions compared to TSP)

Various hazardous air pollutants would be emitted in very small quantities. Total hazardous air pollutants that would be emitted from the proposed activities would be less than 1 kilogram annually, less than 0.01 percent of Aiken County's annual hazardous air pollutants emissions of 1.9 million kilograms.

Nitric acid would be emitted in relatively small quantities as well; approximately 176 kilograms annually, or a daily average of less than 0.5 kilograms. If that amount were distributed evenly in a 1-square mile box up to a mixing height of 3,000 feet, the concentration would be about 10 $\mu g/m^3$, far less than the maximum allowable concentration of 125 $\mu g/m^3$. Because H-Area is approximately 8 miles from the SRS fenceline, the nitric acid concentration at the site boundary would be even lower.

Employee Travel—Full-time employees required for operations under the H-Area alternative would contribute air emissions through commuting in personal vehicles. Estimates of emissions from employees commuting in personal vehicles assumes that each employee travels separately, that 29 percent of the employees travel 100 miles, and 71 percent travel 50 miles (average of

65 vehicle miles traveled). Emissions from employee travel represent 0.12 percent or less of the Aiken County emissions. This estimate is conservatively high because most of the workers would be existing employees whose commuting emissions are already accounted for in the baseline emissions estimates for the region.

Greenhouse Gases-- Combustion of fossil fuels associated with the H-Area alternatives would result in the emission of carbon dioxide, one of the gases that influence global climate change. Maximum annual carbon dioxide emissions under this alternative, for activities including receipt and storage of casks, carbon digestion, uranium processing, and ultimate disposition (liquid processing, cementation, or vitrification) were estimated based on emissions from material processing; fuel use (see Subsection 4.1.7.7.3); electricity use; employee vehicles; and truck shipments of waste and construction materials. Annual CO_{2e} (CO_2 equivalent) emissions would not exceed 25,000 metric tons (27,600 tons). The Council on Environmental Quality (CEQ) has issued draft guidance that recommends that agencies consider 25,000 metric tons of carbon dioxide equivalent emissions on an annual basis as a reference point below which a quantitative analysis of greenhouse gas is not required (CEQ 2014).

The CO_{2e} emissions that would be generated under this alternative would be a marginal increase over the No Action Alternative, and would not substantially increase CO_{2e} emissions or associated climate change impacts. Because of this, further analysis of GHG emissions and their effect on climate are not needed. In addition, because of the relatively short timeframe of this project, the impacts of this project are not expected to be affected by future climate change.

4.1.3.1.2 L-Area Alternative

Construction—Under the L-Area Alternative, approximately 1.7 acres of land would be disturbed for construction of the new storage pads and roadways to store the CASTOR casks. Construction of the storage pads would not be expected to exceed existing permit levels for SRS portable heavy equipment operation (DOE 2014a).

For construction of the carbon digestion and melt and dilute processes at L-Area, less than 1 acre of land would be disturbed. New walls; a sand filter, fan room and stack; and a truck well would be installed. Typical construction equipment would be used, including a diesel- or gas-powered backhoe, front end loader, road grader, crane, forklift, and a variety of trucks. The construction time is estimated to be 4 years, but not all equipment would be operated throughout the duration (DOE 2014a). Construction of the carbon digestion process at L-Area is not expected to exceed permit limits for SRS portable heavy equipment operation. **Table 4-12** shows the estimated annual construction emissions for the L-Area Alternative. Construction emissions would be less than 1 percent of Aiken County emissions.

Table 4-12: Estimated Annual Construction Emissions under the L-Area Alternative

Source	Emissions Estimates (tons/year)					
	VOC	**SO_2**	**NO_2**	**CO**	**PM_{10}**	**$PM_{2.5}$**
Storage Pad/Roadway Construction Fugitive Emissions	1.1	0.01	7.1	5.9	5.5	0.39
Construction of New Sand Filter, Fan Room, Stack and Truck Well Emissions	0.46	0.01	2.9	2.3	3.6	0.15
Total Annual Emissions	1.6	0.02	10	8.2	9.1	0.54
2011 NEI Aiken County[a]	40,128	5,020	7,646	49,790	23,730	7,217
Percentage of Aiken County Emissions	<0.01	<0.01	0.13	0.02	0.02	0.01

CO = carbon monoxide; NEI =National Emissions Inventory NO_2 = nitrogen dioxide; PM_{10} and $PM_{2.5}$ = particulate matter with a diameter of less than or equal to 10 microns and 2.5 microns, respectively; SO_2 = sulfur dioxide; VOC = volatile organic compounds, TSP=total suspended particulates

[a] Source: EPA 2014d, (total countywide PM-10 emissions compared to TSP)

Operations—Under the L-Area Alternative, no changes above normal operations are expected to result from the receipt, storage, and transfer of CASTOR casks. Therefore, no increase in air emissions is expected from this activity (DOE 2014a).

Table 4-13 shows estimated criteria air pollutant emissions under the L-Area Alternative. The highest total emissions would be from nitrogen dioxide, and would represent 2.3 percent of Aiken County emissions for that pollutant. Emissions from the melt and dilute process in L-Area are expected to be similar to those under the Vitrification Option of the H-Area Alternative (DOE 2014a). Therefore, the analysis in this EA assumes the addition of H-Canyon permitted levels of emissions to L-Area emissions in order to estimate impacts. These would be new emissions for L-Area, therefore the Title V Operating Permit (DOE 2007) may require revision. Any permit revisions would need to be approved by the State of South Carolina, ensuring appropriate emissions control technologies are incorporated and no State or Federal emissions limits are exceeded. Hazardous air pollutant emissions are not expected to increase.

Table 4-13: Estimated Annual Criteria Air Pollutant Emissions under the L-Area Alternative

Source	Emissions Estimates (tons/year)							
	VOC	SO_2	NO_2	CO	PM_{10}	$PM_{2.5}$	TSP	CO_2
Carbon Digestion[a]	NA	NA	30	NA	NA	NA	NA	103
Melt and Dilute Process[a]	NA	NA	147	NA	0.32	NA	0.32	NA
Saltstone Production Facility[b]	NA	NA	NA	NA	NA	NA	0.02	NA
TOTAL	0.00	0.00	177	0.00	0.32	0.00	0.34	103
2011 NEI Aiken County[c]	40,128	5,020	7,646	49,790	23,730	7,217	23,730	23,730
Percentage of Aiken Emissions	0.00	0.00	2.3	0.00	0.00	0.00	0.00	0.43

CO = carbon monoxide; NA = not applicable; NEI =National Emissions Inventory; NO_2 = nitrogen dioxide; PM_{10} and $PM_{2.5}$ = particulate matter with a diameter of less than or equal to 10 microns and 2.5 microns, respectively; SO_2 = sulfur dioxide; VOC = volatile organic compounds, TSP=total suspended particulate

a. Source DOE 2014a

b. Source DOE 2007; adjusted for 16 days of operation. The listed operational duration represents the time required to process wastes from activities associated with spent nuclear fuel from Germany through the Saltstone facilities (about 16 days), and not the time required to process all SRS wastes through the Saltstone facilities.

c. Source USEPA 2012

Employee Travel—Employee commuting emissions estimates assume 29 percent of the vehicles travel 100 miles, 71 percent travel 50 miles (average of 65 vehicle miles traveled) and each employee travels separately. Emissions from employee travel represent less than 0.18 percent of the countywide emissions. This estimate is conservatively high because most of the workers would be existing employees whose commuting emissions are already accounted for in the baseline emissions estimates for the region.

Greenhouse Gases-- Combustion of fossil fuels associated with this alternative would result in the emission of carbon dioxide, one of the gases that influence global climate change. Maximum annual carbon dioxide emissions under this alternative, for activities including receipt and storage of casks, carbon digestion, and ultimate disposition (melt and dilute) were estimated based on fuel use; electricity use; employee vehicles; and truck shipments of waste and construction materials. Annual CO_{2e} emissions would be less than 25,000 metric tons (27,558 tons). CEQ has issued draft guidance that recommends that agencies consider 25,000 metric tons of carbon dioxide equivalent emissions on an annual basis as a reference point below which a quantitative analysis of greenhouse gas is not required (CEQ 2014).

The CO_{2e} emissions that would be generated under this alternative would be a marginal increase over the No Action Alternative, and would not substantially increase CO_{2e} emissions or associated climate change impacts. Because of this, further analysis of GHG emissions and their effect on climate are not needed. In addition, because of the relatively short timeframe of this project, the impacts of this project are not expected to be affected by future climate change.

4.1.3.2 Human Health

This section presents radiological impacts on workers and the public from normal operations and postulated accidents at SRS, as well as impacts from possible chemical exposures and accidents and intentional destructive acts.

Health risks are considered for involved and noninvolved workers[34], the offsite population, and a maximally exposed individual (MEI). Workers and members of the public are protected from exposure to radioactive material and hazardous chemicals by facility design and construction and administrative procedures. Major DOE design criteria include those in DOE Order 420.C, "Facility Safety," and DOE Order 430.1B, Change 2, "Real Property Asset Management." DOE regulation 10 CFR Part 830, "Nuclear Safety Management," requires documented safety analyses and technical safety requirements that provide the safety basis and controls for facility design and operation. Other regulations and DOE directives include 10 CFR Part 820, "Procedural Rules for DOE Nuclear Facilities," DOE Order 458.1, "Radiation Protection of the Public and the Environment," 10 CFR Part 835, "Occupational Radiation Protection," and 10 CFR Part 851, "Worker Safety and Health Program."

To protect the public from impacts from radiological exposure, DOE Order 458.1 imposes an annual individual dose limit of 10 millirem from airborne pathways, 100 millirem from all pathways, and 4 millirem from the drinking-water pathway. Public doses from all pathways must be maintained to levels ALARA. To protect workers from impacts from radiological exposure, 10 CFR Part 835 imposes an individual dose limit of 5,000 millirem in a year. In addition, worker doses must be monitored and controlled below the regulatory limit to ensure that individual doses are less than an administrative limit of 2,000 millirem per year, and maintained to ALARA levels. The SRS ALARA goal is to limit annual individual exposures to 500 millirem (SRS 2014).

Nonradiological public health impacts may occur primarily through inhalation of air containing hazardous chemicals that are released to the atmosphere. (Risks from other pathways such as ingestion of contaminated drinking water are generally lower.) Impacts are minimized through design, construction, and administrative controls that decrease hazardous chemical releases to the environment and achieve compliance with permit requirements (e.g., NESHAPs and NPDES permits). The effectiveness of these controls is verified through the use of environmental monitoring information and inspection of mitigation measures.

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[34] An involved worker is directly or indirectly involved with operations at a facility who receives an occupational radiation exposure from direct radiation (i.e., neutron, x-ray, beta, or gamma) or from radionuclides released to the environment from normal operations. A noninvolved worker is a site worker outside of a facility who would not be subject to direct radiation exposure, but could be exposed to emissions from that facility, particularly during postulated accidents. The offsite population comprises members of the general public living within 50 miles (80 kilometers) of a facility. The MEI is a hypothetical member of the public at a location of public access that would result in the highest exposure, which is assumed to be at the SRS boundary during normal operations and postulated accidents (DOE 2015). For individuals or population groups, estimates of potential LCFs are made using a risk estimator of 0.0006 LCFs per rem or person-rem (DOE 2003). For an acute dose to an individual equal to or greater than 20 rem, the factor is doubled (NCRP 1993). An LCF risk to a population represents the estimated number of LCFs within that population and may be larger than 1; an LCF risk to an individual represents the probability of that individual receiving an LCF and is always less than or equal to 1.

Nonradiological impacts on SRS workers could occur through exposure to hazardous materials by inhaling contaminants in the workplace atmosphere or by direct contact. Workers are protected from workplace hazards through appropriate training, protective equipment, monitoring, materials substitution, and engineering and management controls. They are also protected by adherence to Federal and state laws, DOE orders and regulations, and Occupational Safety and Health Administration (OSHA) and EPA guidelines. Monitoring that reflects the frequency and quantity of chemicals used in the operational processes ensure that these standards are not exceeded. DOE requires that conditions in the workplace be as free as possible from recognized hazards that cause, or are likely to cause, illness or physical harm.

4.1.3.2.1 __Normal Operations__

This section summarizes radiological impacts on the public and involved workers from normal operations. Subsequent sections provide more-detailed descriptions of the activities involved in managing the spent nuclear fuel from Germany that contribute to these impacts.

Summary of Radiological Impacts on Members of the Public

Construction or modification of SRS facilities to enable receipt and management of spent nuclear fuel from Germany would not result in impacts on members of the public. Small levels of impacts on members of the public could occur, however, under operations performed in accordance with all action alternatives.

Table 4-14 summarizes annual and life-of-project radiation doses and risks to members of the public under the H- and L-Area action alternatives. Annual doses to the population within 50 miles (80 kilometers) of SRS range from 2.3 to 7.8 person-rem; annual doses to an MEI range from 0.029 to 0.12 millirem. Annual doses were estimated for all alternatives by conservatively assuming concurrent activities at all locations that could result in meaningful impacts on the public. No LCFs are expected among the population or to the MEI.

Summary of Radiological Impacts on Involved Workers

Table 4-15 summarizes annual and life-of-project radiation doses and risks for involved workers due to construction and operational activities under the H- and L-Area action alternatives. The only meaningful construction doses would occur under the H-Area Alternative from H-Canyon modifications to install a carbon digestion capability. The involved worker populations could receive an annual dose of about 17 person-rem and a total dose of about 50 person-rem. No LCFs are expected (calculated annual and total risks are 0.01 LCFs and 0.03 LCFs, respectively). These doses would be received by involved workers under all H-Area Alternative processing options. Construction activities at H-Area under the LEU Waste and LEU/Thorium Waste Options to install a uranium solidification capability would occur outside of radiation control areas. Similarly under the L-Area Alternative, workers would install carbon digestion and melt and dilute capabilities outside of L-Area radiation control areas. For either construction activity, workers are not expected to receive meaningful radiation doses.

Table 4-14: Summary of Radiation Doses and Risks for Members of the Public from Operations at Savannah River Site

Impact Parameter	H-Area Alternative			L-Area Alternative
	Vitrification Option	LEU Waste Option	LEU/Thorium Waste Option	
Population within 50 Miles (80 Kilometers)				
Annual dose (person-rem)	7.3	7.8	7.6	2.3
Percent of natural background radiation[a]	3×10^{-3}	3×10^{-3}	3×10^{-3}	8×10^{-4}
Annual LCFs[b]	0 (4×10^{-3})	0 (5×10^{-3})	0 (5×10^{-3})	0 (1×10^{-3})
Life-of-project LCFs[b]	0 (0.01)	0 (0.01)	0 (0.01)	0 (9×10^{-3})
Maximally Exposed Individual				
Annual dose (millirem)	0.084	0.12	0.12	0.029
Percent of natural background radiation[a]	0.03	0.04	0.04	9×10^{-3}
Annual LCF risk	5×10^{-8}	6×10^{-8}	6×10^{-8}	2×10^{-8}
Life-of-project LCF risk	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}

LCF = latent cancer fatality; LEU = low-enriched uranium.

[a] The annual dose from natural background radiation in the area around SRS is assumed to be 311 millirem for the average individual (NCRP 2009); the population within 50 miles (80 kilometers) of H-Area in 2020 would receive a dose from this background radiation of about 276,000 person-rem.

[b] The reported values are the number of LCFs expected to occur in the 50-mile (80-kilometer) population under any alternative and are presented as whole numbers; the values in parentheses are the statistically calculated values.

Note: Risks were determined using a factor of 0.0006 LCFs per rem or person-rem (DOE 2003) and are presented using one significant figure.

Source: DOE 2014a

During operations, annual doses to the involved worker populations would range from 8.0 to 41 person-rem, while life-of-project doses would range from 43 to 69 person-rem. No LCFs are expected among the involved worker population (calculated annual risks range from 5×10^{-3} to 0.02 and total risks range from 0.03 to 0.04 LCFs).

Table 4-15: Summary of Radiation Doses and Risks for Involved Savannah River Site Workers

Impact Parameter	H-Area Alternative			L-Area Alternative
	Vitrification Option	LEU Waste Option	LEU/Thorium Waste Option	
Construction				
Annual dose (person-rem)	17	17	17	-[b]
Annual risk (LCF)[a]	0 (0.01)	0 (0.01)	0 (0.01)	-[b]
Life-of-Project dose (person-rem)	50	50	50	-[b]
Life-of-Project risk (LCF)[a]	0 (0.03)	0 (0.03)	0 (0.03)	-[b]
Operations				
Annual dose (person-rem)	41	28	28	8.0
Annual risk (LCF)[a,c]	0 (0.02)	0 (0.02)	0 (0.02)	0 (5×10^{-3})
Life-of-Project dose (person-rem)	69	61	61	43
Life-of-Project risk (LCF)[a,c]	0 (0.04)	0 (0.04)	0 (0.04)	0 (0.03)

LCF = latent cancer fatality; LEU = low-enriched uranium.

[a] The reported values are the numbers of LCFs expected to occur in the population and are presented as whole numbers; the values in parentheses are the statistically calculated values.

[b] Because work would not be performed in a radiation area, meaningful radiation doses among involved workers are not expected.

Note: Risks were determined using a factor of 0.0006 LCFs per person-rem and are presented using one significant figure (DOE 2003).

Radiological Impacts by Major SRS Activity Involving Spent Nuclear Fuel from Germany

Doses and risks among members of the public and involved workers were also evaluated as a function of the following major activities at SRS involving spent nuclear fuel from Germany:

- Receipt of CASTOR casks at SRS and storage at H-Area and/or L-Area
- Inspection of stored CASTOR casks
- Transfer of stored CASTOR casks for carbon digestion at H-Area or L-Area
- Digestion of spent nuclear fuel at H-Canyon or L-Area to separate HEU kernels from their carbon matrices
- Processing of HEU kernels at H-Canyon or L-Area
- Disposition of waste generated from spent nuclear fuel storage, digestion, and processing.

Doses and risks to members of the public for each major activity are listed in **Table 4-16**. Doses and risks associated with cask receipt, inspection, and transfer for carbon digestion are not listed in this table because these activities would not involve public doses or risk. Doses and risks received by involved SRS workers are listed in **Table 4-17** for each major activity involving storage and treatment of the spent nuclear fuel at H-Area or L-Area, and in **Table 4-18** for waste management activities involving DWPF and the saltstone facilities[35].

[35] The listed operational years in Tables 4-16 through 4-18 represent the times required to process the spent nuclear fuel from Germany pursuant to each major activity, which may require less than 1 year for some activities. For example, the time required for vitrification of HLW at DWPF under the H-Area Alternative, Vitrification Option, represents only the time required to vitrify the HLW generated from that alternative option (about 100 days), and not the time required to process all SRS HLW at DWPF.

Table 4-16: Radiological Doses and Risks for Members of the Public by Major Operational Activity Involving Spent Nuclear Fuel from Germany

	H-Area Operations (H-Area Alternative)			L-Area Operations (L-Area Alternative)		DWPF Operations			Saltstone Facilities Operations		
		Kernel Processing[a]				H-Area Alternative			H-Area Alternative		
Parameter	**Carbon Digestion**	**Vitrification Option**	**LEU Waste or LEU/Thorium Waste Option**	**Carbon Diges-tion**	**Melt and Dilute**	**Vitrification Option**	**LEU Waste or LEU/Thorium Waste Option**	**L-Area Alternative[b]**	**Vitrification Option**	**LEU Waste or LEU/Thorium Waste Option**	**L-Area Alternative**
Operational years[c]	3.5	2	2	7	7	0.3	0.08	0	0.07	0.08	0.04
Population Within 50 Miles (80 Kilometers)											
Annual dose (person-rem)	4.9	0.26	0.29	2.0	0.20	0.028	8.3×10^{-3}	-	2.1	2.6	0.13
Annual risk (LCFs)[e]	0 (3×10^{-3})	0 (2×10^{-4})	0 (2×10^{-4})	0 (1×10^{-3})	0 (1×10^{-4})	0 (2×10^{-5})	0 (5×10^{-6})	-	0 (1×10^{-3})	0 (2×10^{-3})	0 (8×10^{-5})
Life-of-Project dose (person-rem)	17	0.52	0.57	14	1.4	0.028	8.3×10^{-3}	-	2.1	2.6	0.13
Life-of-Project risk (LCFs)[e]	0 (0.01)	0 (3×10^{-4})	0 (3×10^{-4})	0 (8×10^{-3})	0 (8×10^{-4})	0 (2×10^{-5})	0 ($5 \times 1 0^{-6}$)	-	0 (1×10^{-3})	0 (2×10^{-3})	0 (8×10^{-5})
Maximally Exposed Individual											
Annual dose (millirem)	0.046	0.0024	0.003	0.024	0.0024	4.3×10^{-4}	1.3×10^{-4}	-	0.036	0.047	0.0022
Annual risk (LCF)	3×10^{-8}	1×10^{-9}	2×10^{-9}	1×10^{-8}	1×10^{-9}	3×10^{-10}	8×10^{-11}	-	2×10^{-8}	3×10^{-8}	1×10^{-9}
Life-of-Project dose (millirem)	0.16	0.0048	0.0057	0.17	0.017	4.3×10^{-4}	1.3×10^{-4}	-	0.036	0.047	0.0022
Life-of Project risk (LCF)	1×10^{-7}	3×10^{-9}	3×10^{-9}	1×10^{-7}	1×10^{-8}	3×10^{-10}	8×10^{-11}	-	2×10^{-8}	3×10^{-8}	1×10^{-9}

DWPF = Defense Waste Processing Facility; HLW = high-level radioactive waste; LEU = low-enriched uranium; LCF = latent cancer fatality.

[a] The listed values are for kernel dissolution (Vitrification Option) or kernel dissolution and solvent extraction with subsequent solidification at H-Area of LEU or LEU/thorium solutions (LEU and LEU/Thorium Waste Options). Solidification alone is projected to take 1.5 years.

[b] Because no liquid HLW is expected from melt and dilute activities under the L-Area Alternative, no waste would require vitrification at DWPF.

[c] Indicates the approximate projected time, in years, required to accomplish each major operational activity. For example, it is expected that vitrification of HLW at DWPF under the H-Area Alternative, Vitrification Option, would require an additional 100 days of DWPF operation, or about 0.3 year.

[d] Impacts for the vitrification option reflect those for all activities at H-Canyon/HB-Line involving nuclear material, not just those related to processing the spent nuclear fuel from Germany kernels.

[e] The reported values are the numbers of LCFs expected to occur in the population and are presented as whole numbers; the value in parentheses are the statistically calculated values.

Note: Risks were determined using a factor of 0.0006 LCFs per rem or person-rem and are presented using one significant figure (DOE 2003).

Table 4-17: Involved Worker Radiation Doses and Risks from Receipt, Storage, and Processing Spent Nuclear Fuel from Germany at H-Area or L-Area

Parameter	Receive Casks at SRS	Inspect Stored Casks	Transfer Casks to H-Area or L-Area	H-Area Operations (H-Area Alternative)			L-Area Operations (L-Area Alternative)	
				Carbon Digestion of Kernels	Kernel Processing Option[a]		Carbon Digestion of Kernels	Melt and Dilute Kernels
					Vitrification	LEU Waste or LEU/Thorium Waste		
Operational years[b]	3.5	9	3.5	3.5	2	1.5	7	7
Annual dose (person-rem)	2.0	1.5	2.2	2.2	-	10	1.1	1.1
Annual risk (LCF)[c]	0 (1×10^{-3})	0 (9×10^{-4})	0 (1×10^{-3})	0 (1×10^{-3})	-	0 (6×10^{-3})	0 (6×10^{-4})	0 (6×10^{-4})
Life-of-Project dose (person-rem)	7.3	13	7.6	7.6	-	15	7.6	7.6
Life-of-Project risk (LCF)[c]	0 (4×10^{-3})	0 (8×10^{-3})	0 (5×10^{-3})	0 (5×10^{-3})	-	0 (9×10^{-3})	0 (5×10^{-3})	0 (5×10^{-3})

LCF = latent cancer fatality; LEU = low-enriched uranium.

[a] No radiation doses are expected among involved H-Canyon workers performing fuel dissolution (Vitrification Option) or fuel dissolution and solvent extraction operations (LEU Waste or LEU/Thorium Waste Options) beyond those normally experienced at H-Canyon. The listed operational years, doses and risks under the LEU Waste or LEU/Thorium Waste Options are for solidification of separated uranium or uranium and thorium.

[b] Indicates the approximate projected time, in years, required to accomplish each major operational activity. For example, it is expected that carbon digestion of kernels under the H-Area Alternative would require approximately 3.5 years to complete.

[c] The reported values are the numbers of LCFs expected to occur in the involved worker population and are presented as a whole numbers; the values in parentheses are the statistically calculated values

Note: Risks were determined assuming 0.0006 LCFs per person-rem and presented using one significant figure (DOE 2003).

Source: DOE 2014a.

Table 4-18: Involved Worker Radiation Exposures from Processing Waste at DWPF and the Saltstone Facilities

Parameter	DWPF Operation			Saltstone Operation		
	H-Area Alternative		L-Area Alternative[a]	H-Area Alternative		L-Area Alternative
	Vitrification Option	LEU Waste or LEU/Thorium Waste Option		Vitrification Option	LEU Waste or LEU/Thorium Waste Option	
Operational years[b]	0.3	0.08	-	0.07	0.08	0.04
Annual dose (person-rem)	33	9.9	-	0.31	0.39	0.21
Annual risk (LCF)[c]	0.02	6×10^{-3}	-	2×10^{-4}	2×10^{-4}	1×10^{-4}
Life-of-Project dose (person-rem)	33	9.9	-	0.31	0.39	0.21
Life-of-Project risk (LCF)[c]	0.02	6×10^{-3}	-	2×10^{-4}	2×10^{-4}	1×10^{-4}

DWPF = Defense Waste Processing Facility; HLW – high-level radioactive waste; LCF = latent cancer fatality; LEU = low-enriched uranium.

[a] Under the L-Area Alternative, carbon digestion and melt and dilute operations would generate low-activity liquid waste which would not require vitrification at DWPF, but would be dispositioned at the saltstone facilities.

[b] Indicates the approximate projected time, in years, required to accomplish each major operational activity. For example, it is expected that vitrification of HLW at DWPF under the H-Area Alternative, Vitrification Option, would require an additional 100 days of DWPF operation, or about 0.3 year.

[c] The reported values are the number of LCFs expected among the involved worker population under any alternative and are reported as whole numbers; the values in parentheses are the statistically calculated values.

NOTE: Risks were determined using a factor of 0.0006 LCFs per person-rem and presented using one significant figure (DOE 2003).

Source: DOE 2001a, 2014a; 2015.

Receipt and Storage of CASTOR Casks

Construction - Modifications to H-Area and/or L-Area to facilitate storage of spent nuclear fuel would occur under both the H-Area and L-Area Alternatives, and would occur outside of current radiation areas. No radiological air emissions are expected in excess of those from normal maintenance activities, and no emissions to the air or releases to ground or surface water pathways are expected that would result in radiological doses to members of the public. No radiological doses are expected among involved construction workers.

Operations – Under both the H-Area and L-Area Alternatives, no radiological releases would be expected from the casks. Although the casks would emit ionizing radiation, members of the public would be excluded from cask receipt and storage areas. Therefore, there would be no radiological impacts on the public during cask receipt and storage.

Cask receipt and storage could result in radiation exposures to involved workers. Receipt and transfer of the casks to storage locations would require less than ten workers per cask, including riggers, drivers, crane operators, and supervisors, but only riggers (assumed to be four) would work close enough to the casks to receive measurable radiation doses. Assuming a dose rate at the cask surfaces of 1 millirem per hour and a 4-hour handling period for each cask, the maximum exposure received by a worker would be about 4 millirem per cask and the total crew dose would be about 0.016 person-rem per cask (DOE 2014a). Table 4-17 shows the involved worker impacts on an annual basis and for the duration of this activity. The annual dose to the involved worker population would be about 2.0 person-rem and the total dose, assuming receipt of all 455 casks would be 7.3 person-rem. No LCFs would be expected among the workers, with a calculated LCF value of 1×10^{-3} from the annual dose and 4×10^{-3} from the total dose.

Once casks are in storage, involved workers could receive radiation doses during daily inspections. The radiation levels at the storage location and radiation doses received by workers would increase as casks are received and stored, and decrease as casks are transferred for spent nuclear fuel processing. An average daily exposure of 0.004 person-rem is assumed (DOE 2014a); this would result in an annual dose of about 1.5 person-rem and a total dose, assuming a 9-year cask storage period, of about 13 person-rem. No LCFs would be expected among the workers, with calculated LCF values of 9×10^{-4} from the annual dose and 8×10^{-3} from the total dose.

Transfer of Casks for Carbon Digestion

Construction – Under both the H-Area and L-Area Alternatives, no facility construction or modification would be required to enable transfer of casks from storage areas to carbon digestion operations at H-Canyon or L-Area; hence, there would be no impacts among members of the public or workers.

Operations – Under both the H-Area and L-Area Alternatives, no impacts among members of the public would result from transfer of casks from storage areas in H-Area or L-Area to locations in H-Area or L-Area where carbon digestion would occur. There would be no radiological releases and members of the public would be excluded from cask transfer activities and, therefore, would receive no radiation exposure.

Cask transfer could result in radiation exposures to involved workers. Cask transfer could occur by rail car or onsite roads using a transporter. By either method, fewer than ten workers would be involved for each cask transfer; drivers, crane operators, and supervisors would work at a distance from the casks and thus receive little exposure. Riggers could receive a measurable dose from working in closer proximity to the casks; assuming four riggers per cask, the total worker dose during transfer of a single cask would be about 0.016 person-rem (DOE 2014a).

It is assumed that cask transfer would proceed as needed to feed carbon digestion activities at H-Canyon or L-Area, that is, up to 135 casks per year. Table 4-17 lists involved worker doses on an annual basis (2.2 person-rem) and over the approximately 3.5-year duration of this activity (7.6 person-rem). The most likely result would be no LCFs among the workers based on calculated values of 1×10^{-3} from the annual dose and 5×10^{-3} from the total dose to the involved worker population.

Carbon Digestion

Construction – Under the H-Area Alternative, modifications to H-Canyon to provide a carbon digestion capability would require about 3 years to complete, and are not expected to result in emissions to air or water in excess of those from normal H-Canyon operations (which include facility maintenance and equipment replacement and upgrade) (DOE 2014a). Hence, the modifications would not result in incremental impacts on members of the public.

H-Canyon modifications would result in an annual dose to involved workers of about 17 person-rem and a total dose of about 50 person-rem (DOE 2014a). The most likely result would be no LCFs among the workers based on calculated LCF values of 0.01 from the annual dose and 0.03 from the total dose.

Under the L-Area Alternative, construction and modification activities at L-Area would be remotely performed in the facility's hot cell area. Consequently, no meaningful radiological impacts are expected on involved workers.

Operations – Under the H-Area Alternative, digestion of graphite from spent nuclear fuel could result in increased emissions of tritium, carbon-14, chlorine-36, cesium-137, iodine-129, and krypton-85 to the air compared to those from current H-Canyon operations. As shown in Table 4-16, carbon digestion in H-Area would result in an annual population dose of 4.9 person-rem and an annual MEI dose of 0.046 millirem (SRNL 2014b). Based on these doses and a projected activity duration of approximately 3.5 years, no LCFs are expected among members of the public within 50 miles (80 kilometers) of H-Area, with calculated LCF values of 3×10^{-3} from the annual dose and 0.01 from the total dose from carbon digestion. The annual risk of an LCF to the MEI would be 3×10^{-8} and the life-of-project risk would be 1×10^{-7}.

Under the L-Area Alternative, the proposed air treatment system would be similar to that for carbon digestion at H-Canyon with emissions of the same radionuclides to the air. Annual emissions would be half of those for H-Area because the quantity of material annually processed at L-Area would be half that for H-Area. The stack characteristics would also be different – for example, the L-Area stack would be shorter than the H-Area stack. As shown in Table 4-16, carbon digestion at L-Area would result in an annual population dose of 2.0 person-rem and an annual MEI dose of 0.024 millirem (SRNL 2014b, 2014c). Considering these doses and a projected activity duration of 7 years, no LCFs are expected among members of the public with calculated LCF values of 1×10^{-3} from the annual dose and 8×10^{-3} from the total dose for this activity. The annual risk of an LCF to the MEI would be 1×10^{-8}; the total LCF risk to the MEI would be 1×10^{-7}.

Under both the H-Area and L-Area Alternatives, exposures to involved workers would result primarily from preparation of casks for removal of inner canisters containing the spent nuclear fuel (e.g., removing the cask double lid system), and from decontamination of the casks as needed for their disposition (DOE 2014a). Removal of the inner canisters from the casks and subsequent carbon digestion would be performed remotely with minimal additional exposures expected.

Based on a dose rate on contact with the casks of about 1 millirem per hour, the maximum work crew exposure from preparing a cask for removal of the canister would be less than 0.016 person-rem. Under the H-Area Alternative, spent nuclear fuel would be removed from up to 135 casks per year for carbon digestion. As shown in Table 4-17 for "Carbon Digestion of Kernels," involved workers would receive an annual dose of about 2.2 person-rem and a total dose of 7.6 person-rem. No LCFs among the involved worker population are expected based on calculated LCF values of 1×10^{-3} from the annual dose and 5×10^{-3} from the total dose. Under the L-Area Alternative, spent nuclear fuel would be removed from the casks at about half the annual rate as that at H-Canyon, so that involved workers would receive an annual dose of about 1.1 person-rem and the same total dose of about 7.6 person-rem. No LCFs among the involved worker population are expected based on calculated LCF values of 6×10^{-4} from the annual dose and 5×10^{-3} from the total dose.

Processing HEU Kernels

Construction – Under the H-Area Alternative, Vitrification Option, no modifications to H-Area facilities or capabilities would be required. Hence, there would be no impacts on members of the public or involved workers.

Under the H-Area Alternative, LEU Waste and LEU/Thorium Waste Options, minor modifications would be needed within H-Canyon and more extensive construction would be required external to H-Canyon, but within H-Area. Construction of the uranium solidification facility would be external to H-Canyon, and outside of the radiation area at H-Area. Therefore, facility construction and modification activities under this option would not be expected to release radioactive material to air or water that could cause radiation exposures to members of the public.

Involved worker exposures could occur during the minor modifications to H-Canyon, but these exposures would not be expected to add appreciably to the exposures that workers receive as part of normal maintenance activities. No worker exposures would be expected for other construction activities at H-Area.

Under the L-Area Alternative, installation of a melt and dilute capability would occur concurrently with that for installation of a carbon digestion capability. As indicated previously, construction work is not expected to result in meaningful radiation doses among involved workers, and therefore, no LCFs.

Operations – Under the H-Area Alternative, Vitrification Option, no appreciable change would be expected in emissions to air or water compared to those from recent H-Canyon operations involving dissolution and processing of spent nuclear fuel. Hence, no incremental impacts among members of the public would be expected compared to those from recent H-Canyon operations (see Chapter 3, Table 3-7) (DOE 2014a).

For purposes of analysis, it is assumed that dissolving the HEU kernels at H-Canyon, neutralizing the dissolved solutions, and transferring solutions to an SRS tank farm under the Vitrification Option would occur over about 2 years. Annual and total impacts on members of the public from dissolving HEU kernels at H-Canyon are shown in Table 4-16. The impacts reflect those from all activities at H-Canyon/HB-Line involving nuclear material, not just those related to processing the spent nuclear fuel from Germany kernels. No LCFs are expected among members of the public on an annual or life-of-project basis. H-Canyon operations and discharge to the waste system would result in an annual population dose of 0.26 person-rem and an annual MEI dose of 2.4×10^{-3} millirem (DOE 2015a). Based on the population dose and a projected activity duration of 2 years, no LCFs are expected among members of the public with calculated LCF values of 2×10^{-4} from the annual dose and 3×10^{-4} from the total dose. The annual risk of an LCF to the MEI would be 1×10^{-9}; the total LCF risk would be 3×10^{-9}.

Under the H-Area Alternative, LEU and LEU/Thorium Waste Options, activities at H-Canyon would include a dissolution step followed by solvent extraction; these activities would not result in meaningful changes in emissions to air or water compared to those from current and past H-Canyon operations. The annual and life-of-project impacts on the public would be the same for these activities (alone) as those listed in Table 4-16 under the Vitrification Option. To reduce potential radiation doses to involved workers, solidification of the uranium or uranium/thorium solution would occur as soon as reasonably practical after the solvent extraction process. An LEU or LEU/thorium solidification process in H-Area (uranium solidification facility) would operate for 1.5 years.

DOE used the design and operating parameters for the Waste Solidification Building in F-Area at SRS (DOE 2015a) to estimate the impacts of operation of the uranium solidification facility. DOE believes that annual impacts on members of the public would be less than those from operation of the Waste Solidification Building because the uranium solidification facility would have a smaller throughput of a less radiotoxic material than that for the Waste Solidification Building. Processing operations at H-Area and the uranium solidification facility are estimated to result in a combined dose to the public within 50 miles (80 kilometers) of H-Area of 0.29 person-rem and an annual dose to the MEI of 0.0030 millirem. Based on the population dose and the projected activity durations (2 years for H-Canyon and 1.5 years for LEU or LEU/thorium solidification), no LCFs are expected among members of the public, with calculated LCF values of 2×10^{-4} from the annual dose and 3×10^{-4} from the total dose. The annual risk of an LCF to the MEI would be 2×10^{-9}; the total LCF risk would be 3×10^{-9}.

Consistent with the analysis for the *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE/EIS-0306) (DOE 2000a), it is assumed for this Draft EA that melt and dilute operations at L-Area would release all tritium and noble gases that remained in the HEU kernels following carbon digestion and that these radionuclides would be all discharged from the L-Area stack. As shown in Table 4-16, melt and dilute operations at L-Area would thus result in an annual population dose of 0.20 person-rem and an annual MEI dose of 0.0024 millirem (SRNL 2014b, 2014c). Based on the population dose and the projected activity duration (approximately 7 years), no LCFs are expected among members of the public, with calculated LCF values of 1×10^{-4} from the annual dose and 8×10^{-4} from the total dose. The annual risk of an LCF to the MEI would be 1×10^{-9}; the total LCF risk would be 1×10^{-8}.

For any of the processing options, H-Canyon operations are not expected to be different from recent operations, with no changes expected in basic H-Canyon radiation exposure levels or the numbers of exposed workers (DOE 2014a). Therefore, no increase in radiation exposures to H-Canyon involved workers would be expected compared with those from recent H-Canyon operations. Additional worker exposures could occur at the uranium solidification facility. As shown in Table 4-17, these exposures could result in an annual radiation dose of about 10 person-rem (DOE 2014a). Over an estimated 1.5 years of operation, the total worker dose would be 15 person-rem. No LCFs would be expected among the involved worker population on an annual (calculated value of 6×10^{-3}) or on a total activity basis (calculated value of 9×10^{-3}).

Solidified LEU or LEU/thorium could require temporary storage at H-Area or E-Area before disposition. Storage of this waste would occur within CASTOR casks under the HEU Waste Option or a mixture of CASTOR casks and other overpacks (e.g., concrete culvert sections) under the HEU/Thorium Waste Option (See Section 4.1.3.4). There would be no release of radioactive or chemical constituents to the environment during storage and, thus, no impacts on members of the public. No meaningful radiation doses would be expected among workers because the waste would be placed within storage configurations (casks or other overpacks) that would provide for shielding against external radiation.

Under the L-Area Alternative, melt and dilute activities would occur concurrently with carbon digestion activities. Annual and total doses and risks among involved workers would be the same as those from carbon digestion.

Waste Disposition

Construction – Disposition of liquid HLW from the activities proposed in this EA would not require modifications to the existing tank farms, pretreatment infrastructure, DWPF, or the glass waste storage facilities. No modifications would be required to the saltstone facilities in Z-Area; nor would construction of additional capacity at E-Area for disposal of low-level radioactive waste (LLW) or storage of hazardous waste be required. Therefore, there would be no radiological releases from construction activities at these facilities and no radiation exposures to members of the public or SRS workers.

Under the L-Area Alternative, any construction to prepare pads for the concrete overpacks in which the MCOs filled with ingots would be stored would occur outside of L-Area radiation areas. Therefore, there would be no radiological releases from construction activities and no radiation exposures to members of the public or SRS workers.

Operations – Under all processing options for the H-Area Alternative, no modifications would be expected at the tank farm infrastructure or DWPF in S-Area, nor modification or addition of a glass waste storage building or a saltstone disposal unit. It is expected that there would be no additional annual emissions to air or discharge to water from operation of DWPF, the future Salt Waste Processing Facility, and the saltstone facilities in Z-Area (DOE 2014a). Therefore, no meaningful changes would be expected in operations at the tank farms, the HLW pretreatment infrastructure, DWPF and the glass waste storage buildings, and the saltstone facilities, with no changes in annual radiation doses expected among members of the public or workers. However, members of the public would be exposed to emissions from these facilities for the additional periods of time. Under the Vitrification Option, operation of DWPF would be extended by approximately 100 days (0.3 years) and operation of the saltstone facilities and future Salt Waste Processing Facility would be extended by approximately 24 days (0.07 years). Under the LEU Waste and LEU/Thorium Waste Options, operation of DWPF, the future Salt Waste Processing Facility, and the saltstone facilities would each be extended by approximately 30 days (0.08 years) (DOE 2014a).

Under the L-Area Alternative, no HLW would be generated that would require vitrification at DWPF. The MCOs containing the aluminum-uranium-thorium ingots would be transferred to and stored within concrete storage overpacks on an L-Area pad. There would be no gaseous emissions or liquid effluents from this activity and, consequently, no radiation doses would be received by members of the public. Some radiation doses could be received by workers involved in MCO transfer and subsequent inspection activities pending the ultimate disposition of the MCOs. As with all activities associated with management of the spent nuclear fuel from Germany, involved workers would be monitored and radiation doses would be controlled below the regulatory limit to ensure that individual doses are less than an administrative limit of 2,000 millirem per year, and maintained to ALARA levels. The SRS ALARA goal is to limit annual individual exposures to 500 millirem (SRS 2014).

Under the L-Area Alternative, the low-activity radioactive liquids from the melt and dilute process would be combined with grout at the Saltstone Production Facility and disposed of at the Saltstone Disposal Facility. Operation of the saltstone facilities under this Alternative would be extended by approximately 16 days (0.04 years).

Impacts on the public from operation of DWPF and the saltstone facilities are scaled from estimates in previous NEPA documents. For operation of DWPF, the *Final Supplemental Environmental Impact Statement Defense Waste Processing Facility* (*DWPF SEIS*) (DOE 1994b) estimated an annual MEI dose of 1.1×10^{-3} millirem and an annual population dose of 0.071 person-rem. For operation of the saltstone facilities, including salt waste pretreatment, the *Savannah River Site Salt Processing Alternatives Final Supplemental Environmental Impact Statement* (*SPA SEIS*) (DOE 2001a) estimated an annual MEI dose of 0.40 millirem and an annual population dose of 22 person-rem. Operation of these facilities to process waste under the action alternatives would result in public doses that are a fraction of those estimated in the DWPF SEIS and SPA SEIS analyses. The radiation doses to the population within a 50-mile (80-kilometer) radius of SRS were estimated in the *DWPF EIS* and *SPA SEIS* (DOE 1994b, 2001a) assuming a population of 620,000 using census data for 1990. If the doses were scaled to a more recent projection of SRS-area population of 886,000 by the year 2020 (DOE 2015a), the population doses would increase by a factor of approximately 1.4.

Under the H-Area Alternative, Vitrification Option, DWPF processing would result in a population dose of 0.028 person-rem and an MEI dose of 4.3×10^{-4} millirem. Saltstone processing would result in a population dose of 2.1 person-rem and an MEI dose of 0.036 millirem. Under the LEU Waste or the LEU/Thorium Waste Option, DWPF processing would result in a population dose of 8.3×10^{-3} person-rem and an MEI dose of 1.3×10^{-4} millirem. Saltstone processing would result in a population dose of 2.6 person-rem and an MEI dose of 0.047 millirem. Under the L-Area Alternative, saltstone processing would result in a population dose of 0.13 person-rem and an MEI dose of 0.0022 millirem. As summarized in Table 4-16, no LCFs would occur among the 50-mile

(80-kilometer) population under any alternative. Under any alternative, the life-of-project risk of an LCF to the MEI would be no larger than 3×10^{-10} from DWPF operations and no larger than 3×10^{-8} from saltstone operations.

Involved worker impacts from the additional days of operation of DWPF and the saltstone facilities are summarized in Table 4-18. DWPF impacts were scaled from information in the *Final Surplus Plutonium Disposition Supplemental Environmental Impact Statement* (*SPD Supplemental EIS*), assuming 500 involved workers, each having an average annual dose of 0.24 person-rem (DOE 2015a). Worker impacts for the saltstone facilities were estimated using projections from the *SPA SEIS* (DOE 2001a), considering activities for production and disposal of grout into saltstone disposal units and for pretreatment of HLW using a solvent extraction capability. No LCFs are expected among the involved worker population under any alternative.

Activities at E-Area in support of the actions and options addressed in this EA are not expected to result in meaningful incremental impacts on members of the public or involved workers from disposal of LLW or staging of LLW for offsite shipment. Members of the public would be excluded from E-Area, where operations would involve handling of containerized wastes. Workers would be protected from excessive exposures to radiation by implementing routine operational measures (e.g., time in a radiation zone, distance from a source of radiation, shielding) and by administrative measures such as monitoring that would ensure compliance with DOE requirements for worker protection as summarized in the opening paragraphs of Section 4.1.3.2.

There would be nothing inherent in any LLW that would be generated under either alternative that would present unique challenges to worker health and safety. As discussed in Section 4.1.3.4, disposal of solidified LEU or LEU/thorium at E-Area would require some additional reviews and possible revisions to design and operation of the disposal units receiving the waste. Waste would be placed in the disposal units using standard methods to maintain worker radiological and physical safety (e.g., using a crane to place the waste into the disposal units).

4.1.3.2.2 Facility Accidents

This section summarizes an evaluation of the potential effects on human health from accidents associated with the processing of the spent nuclear fuel from Germany at facilities at SRS. Because it is early in the decision-making process, detailed safety and accident evaluations of the alternatives and options have not been performed, but existing documented safety analyses (DSAs) and NEPA documents, as well as a preliminary, scoping-level assessments for H-Canyon provide sufficient information to assess potential impacts from postulated accidents. Scoping-level accident scenarios and potential source terms have been developed for the SRS facilities. Where it is reasonable to identify how alternatives or options might change the type of accidents or their magnitude, those changes are identified. For example, differences between accidents and source terms associated with the processing or separation of uranium at H-Canyon for the three options are explicitly identified in the appropriate sections to show how the proposed options and alternatives might change accident risks at a specific facility.

4.1.3.2.2.1 Accident Analysis Approach

Potential accidents that might be applicable to processing the spent nuclear fuel from Germany at SRS would be defined and controlled in facility documentation, such as DSAs, hazard assessments (HAs) and consolidated hazards analysis documents. Using a “what-if” type hazards review process, the potential activities associated with transfer, storage, and processing spent nuclear fuel from Germany were reviewed and accident scenarios developed. Potential accidents include radiological and chemical accidents that have a low frequency of occurrence, but large consequences, and a spectrum of other accidents that have higher frequencies of occurrence and smaller consequences. These accident scenarios include: materials at risk (MAR), release mechanisms, source terms (quantities of hazardous materials released to the environment), and frequency and consequences of the specific accident event. These accident evaluations were reviewed along with the preliminary accident information provided for this proposed project (DOE 2014a) to identify the scoping-level accident scenarios for evaluation in this EA.

4.1.3.2.2.2 Cask Storage Accidents at the SRS

CASTOR casks would be transported to the SRS site and stored on outdoor pads in either H-Area or L-Area. The casks would be protected from the elements by enclosure(s) or individual covers. The fuel would remain in canisters within the CASTOR casks until it is transferred to be processed.

The casks are designed and certified to survive a wide range of transportation accidents, including train or truck impacts, large fuel-fed fires, a significant drop, and water immersion, as well as natural phenomena events (including hurricanes, tornados, floods, lightning, and earthquakes) without releasing their contents. In addition, the spent fuel is also in the form of kernels encapsulated in graphite spheres, each about the size of a tennis ball, so an extremely energetic accident event would be required to cause any release of MAR.

A nearly direct impact from a large aircraft could result in conditions similar to or exceeding those encountered in the most severe transportation accidents with a subsequent long-burning fire, and could threaten the integrity of the CASTOR casks. Based on the CASTOR Safety Analysis Report for Packaging (SARP)(LLNL 2014), a long-burning fire alone, such as one associated with a large fuel-fed fire, would not likely result in a release from the cask. A direct, high-velocity impact from a hardened object such as a jet aircraft part could threaten the integrity of the cask and expose the contents to a fuel-fed fire. This type of accident could be similar to those evaluated in the CASTOR SARP. U.S. and international transportation safety analysis practices for spent nuclear fuel casks identify severe accident situations that could threatened the integrity of a cask and identify release fractions based on the container contents and the severity of the accident. For spent fuel casks, release category 4 would result from the cask being damaged and compromised[36]. 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 that for a release category 5 fire.

The estimated probability of a large jet aircraft crash into a CASTOR storage pad in either H- or L-Area is 1×10^{-7} or less per year (SRNS 2012c). This is based on the area of the storage pad and the overflight frequencies for the SRS. Based on the *FRR SNF EIS* (DOE 1996a), the conditional probability of a crash resulting in a release category of 4, 5, or 6 is estimated to be 1×10^{-4}. The probability of a crash (less than 1×10^{-7} per year) followed by a release following the crash (1×10^{-4} per crash) was estimated to be 1×10^{-11} per year. As a result of this low conditional probability, DOE-STD-3014-2006, *Accident Analysis for Aircraft Crash into Hazardous Facilities* (DOE 2006d), does not require additional analysis of a large aircraft crash. Crash of a light aircraft or helicopter would not be expected to threaten the integrity of the CASTOR casks because they do not have sufficient energy or fuel capacity to cause failure.

A *Safety in Design Tailoring Strategy for HTGR Fuel Receipt and Disposition Feasibility Study* prepared by SRNS describes the overall safety approach to be taken for the German fuel receipt and disposition (SRNS 2014d).

If a decision is made to proceed with the proposed action, a detailed consolidated hazards analysis process will be used to identify accident scenarios, assess consequences, and guide development of controls (SRNS 2014d). H-Canyon processes have previously been evaluated to identify and define safety class (SC) and safety significant (SS) systems, structures, and components (SSCs). These SSCs include controls to prevent or mitigate the consequences of accidents, to well below evaluation guidelines. SSCs include the building structure, canyon exhaust ventilation system, sand filter, backup diesel generators, various monitoring, alarm, and interlock systems, and the vessel air purge system (DOE 2014a).

4.1.3.2.2.3 Carbon Digestion, Uranium Kernel Dissolution, and Processing Accidents in H-Canyon under the H-Area Alternative

This section summarizes an evaluation of the potential accidents associated with the spent nuclear fuel from Germany processing under the H-Area Alternative. In order to put the predicted impacts in perspective, they are compared with the accident impacts reported in current safety documents, including the H-Canyon DSA (SRNS 2014a), and other NEPA analyses, including the *FRR SNF EIS* and the *Savannah River Site Spent Nuclear Fuel Final Environmental Impact Statement* (*Spent Nuclear Fuel FEIS*) (DOE 1996a, 2000b).

Accidents associated with the process operations for carbon digestion, dissolution, and the processing options in H-Canyon are expected to fall within the broad categories of accidents identified in the H-Canyon DSA: leak or spill, fire, explosion, criticality, aircraft crash, and natural phenomena events. The H-Canyon DSA (SRNS 2014a) indicates that H-Canyon is a very robust

[36] The *Radiological Consequences of Ship Collisions that Might Occur in U.S. Ports During the Shipment of Foreign Research Reactor Spent Nuclear Fuel to the United States in Break-Bulk Freighters*, (Sandia 1996) presents a scheme for categorizing the severity of accidents and the release of radioactive material from a shipping cask based on the force of the impact and the intensity and duration of a subsequent fire. The larger the number assigned, the lower the probability of the accident and the larger the release of radioactive material.

structure and provides a high degree of inherent confinement, releases from almost all accidents except a beyond-design-basis earthquake would be confined within the structure and would be filtered through the facility's sand filter prior to release to the environment. Of all the accidents considered in the H-Canyon DSA and supporting safety documents, including a beyond-design-basis earthquake, accidents that result in large fires present the greatest potential for release of radionuclides to the environment.

H-Canyon was designed to process very large quantities of spent fuel shortly after removal from a nuclear reactor. As such, it has the engineered controls to ensure safe operations even when processing spent nuclear fuel containing large quantities of volatile radionuclides, fission products, and actinides. In comparison, the AVR/THTR spent fuel has been out of the reactors for many years (more than 20 years) and the short half-lived isotopes have decayed away.

The proposed material processing and throughputs associated with carbon digestion, uranium kernel dissolution, and any of the H-Area Alternative options are not expected to add any new accident types to those previously evaluated for H-Canyon. Accident analyses associated with processing spent fuel and other nuclear materials have been extensively evaluated for the H-Canyon facilities, including in the routinely updated DSA for the H-Canyon facilities, *H-Canyon & Outside Facilities, H-Area, Documented Safety Analysis*, S-DSA-H-0001, Rev. 9 (SRNS 2014a) and in the *Spent Nuclear Fuel FEIS* (DOE 2000b). Prior to implementation of any of the proposed options, detailed hazard and accident analyses would be performed and the safety basis documents associated with the proposed operations would be updated as needed. New controls would be established if needed to ensure the overall accident risks remain within DOE limits.

Savannah River Nuclear Solutions has prepared a "*Preliminary Scoping-Level Hazard Analysis for the Processing of HTGR Pebble Fuel at SRS*" (preliminary Hazard Analysis) (SRNS 2015). This analysis was performed using a graded approach consistent with the preliminary design and process inputs that were available. It is intended to meet the requirements for hazard analysis set forth by DOE-STD-1189-2008 (DOE 2008a) for a conceptual design/process. This preliminary Hazard Analysis identifies hazards associated with the proposed activity and compares the hazardous events to the current facility safety basis. The preliminary Hazard Analysis also documents potential engineering controls and design features, along with their proposed functional classification, that may be needed to protect onsite workers, as well as the public.

The preliminary Hazard Analysis (SRNS 2015) indicates that the potential for accidents and the potential accident consequences for workers and the public from the proposed carbon digestion, dissolution, and solvent extraction activities are likely well within the scope of the accident scenarios, MARs, and consequences evaluated in the H-Canyon DSA (SRNS 2014a) and other existing safety documents. DOE would confirm this during the detailed accident analysis that would be prepared if DOE decides to go forward with this alternative. The H-Canyon DSA and supporting safety documents have evaluated processing of various types of spent nuclear fuel as well as materials containing uranium, plutonium-239, and plutonium-238 materials. Of these materials, plutonium materials, especially plutonium-238 material with a curie content 100 to 1,000 times greater than that of the spent nuclear fuel from Germany, result in the highest releases in several design-basis and beyond-design-basis accidents at H-Canyon (SRNS 2014a).

With the exception of carbon digestion in H-Canyon, similar material handling and process activities have been performed in H-Canyon and controls are in place to ensure they would be conducted safely. Because carbon digestion would occur within the heavily shielded facility, any accidental releases associated with the process would likely be mitigated by existing safety features associated with the facility. Because the radiological inventories of the spent nuclear fuel from Germany are within or comparable to those associated with the current and past operations at H-Canyon, the addition of the carbon digestion activities would not be expected to substantially change the highest-consequence accident scenarios and source terms evaluated in existing H-Canyon safety documents (SRNS 2015). Depending on the final decision and method for processing the spent nuclear fuel from Germany, the years of operation of H-Canyon may be extended. The options utilizing solvent extraction can only be performed at the conclusion of all H-Canyon processing due to the potential cross contamination of LEU with comparatively high concentrations of uranium-232. If processing the spent nuclear fuel from Germany were to extend the years of operation of H‑Canyon, some accident risks associated with H‑Canyon processing would continue for a longer period.

Evaluation of the carbon digestion process in H-Area and L-Area is underway. The *Process Description for Processing of HTGR Pebble Fuel at SRS* (SRNL 2014d) indicates that a criticality accident during carbon digestion is not credible, stating that it should be possible to include engineered controls such that an accidental criticality would be extremely unlikely. A complete, peer-reviewed criticality analysis of the new operation would be conducted before operations would commence.

Although the H-Canyon processing details are still in the study/conceptual design phase, the general quantities of materials that might be at risk and the types of hazards that might result from the proposed operations evaluated in the preliminary Hazards Analysis (SRNS 2015) are known. The preliminary Hazards Analysis identified seven types of events: criticality, fires, explosions, loss of confinement, inadvertent worker radiation exposure, external events, and natural phenomena hazards. For each event type, a conservative evaluation of the consequence was made based on the quantities of MAR and information provided by the early process studies and conceptual design for the handling and processing of the spent nuclear fuel from Germany in H-Canyon. The scenarios with the highest potential releases and consequences were identified as the "bounding" events and were then evaluated in detail. Other scenarios, with lower releases were not evaluated in detail, but were considered to ensure that adequate safety controls would be present. The postulated spent nuclear fuel from Germany scenarios were compared to the bounding scenarios for each event type in the current H-Canyon DSA. Possible control strategies were identified and compared to the existing H-Canyon DSA control strategies to identify impacts, if any, to the current facility safety basis.

The preliminary Hazards Analysis (SRNS 2015) did not identify any unique fire hazards that were worse than those identified in the H-Canyon DSA and concluded that the existing fire control strategy credited in the H-Canyon DSA is sufficient to protect site workers and the public. The preliminary Hazards Analysis did not identify any explosive gases at the pre-conceptual design phase, found no unique explosion hazards that were worse than those identified in the H-Canyon DSA, and concluded that the existing canyon structure and exhaust system are adequate to mitigate consequences. For loss of confinement events, including leaks and spills, the preliminary Hazards Analysis found no unique hazards that were worse than those identified in the H-Canyon DSA and concluded that the existing canyon structure and exhaust system are adequate to mitigate consequences. For external events (including vehicle impacts and aircraft impacts) and for natural phenomena events, the preliminary Hazard Analysis found no unique hazards. The preliminary Hazard Analysis concluded that the current control strategy, which includes the shipping package (CASTOR cask), robust H-Canyon structure, and active canyon exhaust ventilation system, is adequate and that processing the German fuel would not impact the current H-Canyon safety basis.

The H-Canyon DSA (SRNS 2014a), the H-Canyon preliminary scoping-level Hazards Analysis (SRNS 2015), and the *Spent Nuclear Fuel FEIS* (DOE 2000b) identify a range of potential accidents in H-Canyon. The potential source terms and consequences for the postulated high-consequence facility accidents based on the H-Canyon DSA and the *Spent Nuclear Fuel FEIS* are presented in **Table 4-19**. Similar accidents were postulated for the processing of the spent nuclear fuel from Germany based on the DOE Savannah River *Data Call Response for German Fuel Environmental Assessment* (DOE 2014a) prepared in support of this EA and compared to those from the H-Canyon DSA and the *Spent Nuclear Fuel FEIS*.

Table 4-19: Potential Accident Impacts for Processing H-Canyon under the H-Area Alternative

Accident[a]	Source Term[b]	Frequency (per year)	Impacts on Noninvolved Worker: Dose[f,] (rem)	Impacts on Noninvolved Worker: Probability of an LCF[d]	Impacts on MEI at the Site Boundary[c]: Dose[f,] (rem)	Impacts on MEI at the Site Boundary[c]: Probability of an LCF[d]	Impacts on Population within 50 Miles: Dose (person-rem)	Impacts on Population within 50 Miles: LCFs[e]
Criticality								
DSA: Other H-Canyon Missions	1.0×10^{19} fissions	Extremely unlikely	0.034	2×10^{-5}	0.0028	2×10^{-6}	1.3	0 (0.0008)
SNF from Germany	Same as above	Same as above	Impacts would be the same as those above[g].					
Leaks and Spills								
High-Consequence *Spent Nuclear Fuel FEIS* processing accident: Processing phase in canyon (coil and tube failure)	Various isotopes	Extremely unlikely	13	8×10^{-3}	1.3	8×10^{-4}	78,000	47
High-Consequence SNF from Germany Accident	Various isotopes	Same as above	Impacts are assumed to be similar to those above[g].					
Fires:								
DSA: Cask car fire while storing or transporting fuel (Event FR-1-002)	Various isotopes	Extremely unlikely	21.6	0.03	0.20	1×10^{-4}	Population impacts are not evaluated in the DSA, consistent with DOE regulations.	
SNF from Germany – LEU Waste and LEU/Thorium Waste - Options: Fire in H-Canyon solvent extraction	Various isotopes	Same as above	Impacts are assumed to be similar to those above.[g]					
Explosions								
Highest-consequence from DSA: Hydrogen explosion in H-Canyon high activity waste container	0.12 Ci Cs-137, 0.0396 Ci Sr-90 and Y-90, and 0.00183 Ci Pu-238	Extremely unlikely	7.0	4×10^{-3}	0.22	1×10^{-4}	Not evaluated in the DSA, consistent with DOE regulations.	
High-Consequence SNF from Germany Accident	Various isotopes	Same as above	Impacts are assumed to be similar to those above.[g]					
Design-Basis Earthquake								
DSA: Design-basis earthquake with fire (H-Canyon)	Various isotopes	Extremely unlikely	15	0.01	0.41	2×10^{-4}	Not evaluated in the DSA, consistent with DOE regulations.	
SNF from Germany contribution: Design-basis earthquake with fire	Various isotopes	Same as above	Impacts are assumed to be similar to those above[g].					

Accident[a]	Source Term[b]	Frequency (per year)	Impacts on Noninvolved Worker: Dose[f,] (rem)	Impacts on Noninvolved Worker: Probability of an LCF[d]	Impacts on MEI at the Site Boundary[c]: Dose[f,] (rem)	Impacts on MEI at the Site Boundary[c]: Probability of an LCF[d]	Impacts on Population within 50 Miles: Dose (person-rem)	Impacts on Population within 50 Miles: LCFs[e]
Beyond-Design-Basis Earthquake								
DSA: Beyond-design-basis earthquake with fire	Various isotopes, unmitigated release	Beyond extremely unlikely	4,000	fatality	164	0.1	Not evaluated in the DSA, consistent with DOE regulations	
SNF from Germany contribution: Beyond design-basis earthquake with fire	Assumed to be same as above	Same as above	Impacts are assumed to be similar to those as above[g].					

DSA=documented safety analysis; LCF = latent cancer fatality; MEI = maximally exposed individual; SNF = spent nuclear fuel.

[a] The scenarios and source terms for non-spent nuclear fuel from Germany processing activities were taken from the highest consequence accidents in the H-Canyon DSA (SRNS 2014a), the H-Canyon preliminary scoping-level Hazards Analysis (SRNS 2015), or the *Spent Nuclear Fuel FEIS* (DOE 2000b).

[b] Source terms for spent nuclear fuel from Germany were calculated using scoping-level scenarios and preliminary source terms

[c] A site boundary distance of 7.3 miles was used.

[d] For hypothetical individual doses equal to or greater than 20 rem, the probability of a LCF was doubled; doses ≥600 rem are assumed to result in a near-term fatality.

[e] The reported values are the numbers of LCFs expected to occur in the population and are presented as whole numbers; the statistically calculated values are provided in parentheses when the reported result is 1 or less.

[f] Doses reported for the Noninvolved Worker and MEI for the DSA events are from the H-Canyon DSA (SRNS 2014a)

[g] The Hazards Analysis team (SRNS 2015) found no unique hazards associated with the processing of the pebbles associated with spent nuclear fuel from Germany in H-Canyon. The projected impacts from processing pebbles in H-Canyon were estimated to be similar to and no greater than those evaluated in the H-Canyon DSA. The potential radiological impacts of the spent nuclear fuel from Germany were compared to the fuel mixtures in the DSA and in the cited EISs and found to present no greater hazard.

Note: To convert grams to ounces, multiply by 0.035274; miles to kilometers, by 1.6093.

Source: DOE 2000b; SRNS 2014a, SRNS 2015

4.1.3.2.2.4 Down Blending Accidents under the H-Area Alternative

Under the LEU Waste and LEU/Thorium Waste Options, the solvent stream containing the uranium or uranium and thorium would be processed to an aqueous stream for down blending. The aqueous stream would then be down blended in existing down-blending tanks in the outside processing area immediately adjacent to the H-Canyon structure called “A-Line”. The “A-Line” processing area is a portion of the H-Canyon processing area called the H-Canyon Outside Facilities.

The principal accidents associated with down blending operations are leaks and spills due to human failure, transfer errors, equipment failure, and external events, such as truck impacts. Other types of accidents, such as explosions, fires, and natural phenomena, such as earthquakes, would be much less likely. With operations outside the shielding provided by the H-Canyon structure, the addition of uranium-233, uranium-232 and its daughters (LEU Waste and LEU/Thorium Waste Options), and thorium (LEU/Thorium Option), may add potential new safety concerns and a new radiological hazard associated with direct radiation exposure to workers or the accidental release and inhalation of these radionuclides. The plan to mitigate these hazards is to process the materials to a solid waste form promptly after they have been extracted in H-Canyon and before substantial ingrowth of the high-activity daughter products. Spent fuel containing uranium-233 has been processed in H-Canyon in the past. For example, a campaign to dissolve Sodium Reactor Experiment fuel, a uranium-thorium alloy fuel with high uranium-233 content, was completed in 2014 (DOE 2014c).

The highest-consequence spill accident for down-blending operations would be the rupture of the largest vessel, with its contents spilling into the containment basin under the tank. A small fraction of the spilled material would be expected to become aerosolized due to the impact forces of the spill and subsequent evaporation (SRNS 2014a). This is a standard type of accident postulated for the H-Canyon Outside Facilities, and controls would be in place to ensure that very little material would be aerosolized. This type of accident was considered in the H-Area DSA (SRNS 2014a) and assigned a “negligible” consequence for facility workers, collocated workers, and members of the public. Within the consolidated hazards analysis process, a “negligible” consequence classification indicates that the impacts to a worker must be less than 5 rem and the impacts to a member of the public must be less than 0.5 rem.

Because this accident is considered a very low-risk accident, the specific impacts have not been calculated. The spill accident is in the “negligible” category in the H-Area DSA, implying that the doses would be less than 5 rem to a noninvolved worker and less than 0.5 rem to the MEI (SRNS 2014a). A noninvolved worker dose of 5 rem represents an LCF risk of 3×10^{-3}. An MEI dose of 0.5 rem represents an LCF risk of 3×10^{-4}.

4.1.3.2.2.5 Cementation Activity Accidents under the H-Area Alternative

A new solidification process (grouting or cementation) would be installed in H-Area to process the down-blended solution into a solid LLW. The highest-consequence accident for the cementation operations in H-Area is expected to be the rupture of the largest vessel, with its contents spilling into the containment basin under the tank. Since the material involved would be cement, the fraction that would become airborne would be less (about an order of magnitude) than a comparable spill involving liquid such as in the down-blending operations discussed for A-Line, H-Canyon Outside Facilities (SRNS 2014a). As with the spill in the previous down blending discussion, this type of accident was considered in the consolidated hazards analysis (SRNS 2014a) and assigned a "negligible" consequence for facility workers, collocated workers, and members of the public.

Because this accident is considered a very low risk accident, the specific impacts have not been calculated, but were considered to fall in the "negligible" category in the consolidated hazards analysis, implying that the doses would be less than 5 rem to a noninvolved worker and less than 0.5 rem to the MEI. A noninvolved worker dose of 5 rem represents an LCF risk of 3×10^{-3}. An MEI dose of 0.5 rem represents an LCF risk of 3×10^{-4}.

4.1.3.2.2.6 Carbon Digestion, Melt and Dilute, Processing and Disposition Accidents under the L-Area Alternative

Carbon Digestion: Under the L-Area Alternative, the L-Area Purification Hot Cell Facility would be modified to provide carbon digestion and melt and dilute capabilities. Additional construction would take place external to the hot cell and would include addition of a sand filter, fan room, stack, and new truck bay. Melt and dilute activities would occur concurrently with carbon digestion activities. The projected air treatment system would be similar to that for carbon digestion at H-Canyon with emissions of the same radionuclides to the air. Specific accident analyses for carbon digestion in the L-Area Purification Hot Cell Facility have not been performed. Because the same general processing activities would occur in L-Area as in H-Canyon, the general process-related accident scenarios that would be associated with carbon digestion in L-Area are expected to be similar to those for the same process in H-Canyon. For some area-wide events, such as a major seismic event, there could be radiological releases of other materials from ongoing L-Area operations as well as those associated with carbon digestion. At this early point in the conceptual design, these differences are unknown, but expected to be within the bounds of the existing L-Area safety basis.

The MAR during carbon digestion in L-Area could be about half of that for H-Area because the quantity of material annually processed at L-Area would be about half that for H-Area. The stack characteristics would also be different – the L-Area stack would be shorter (36 meters) than the H-Area stack (61 meters). As presented in **Table 4-20**, the general accident scenarios associated with carbon digestion at L-Area are the same as those projected for H-Canyon, although the MAR and dose associated with a release could be different due to differences in release height and distance to the SRS boundary and offsite population. These differences are expected to be relatively small such that the impacts from carbon digestion at either H- or L-Area would be similar.

As with carbon digestion in H-Area, the highest-consequence operational accident for carbon digestion in L-Area would be a major fire in a process cell during carbon digestion. As with H-Canyon processing, other accidents associated with the cask cars and seismic events could also be present. The highest-consequence facility-wide accident in L-Area would be an accident for which building confinement is severely degraded, such as a beyond-design-basis earthquake. In spite of the differences in MAR, facility differences, differences in confinement systems and location, the carbon-digestion radiological impacts to the noninvolved workers, the MEI, and the offsite population from accidents associated with carbon digestion in the L-Area Purification Hot Cell are expected to be similar to those projected for carbon digestion operations in H-Area, and no greater than the projected radiological impacts of H-Canyon operation.

Table 4-20: Potential Accident Impacts for Processing under the L-Area Alternative

Accident	Frequency (per year)	Impacts on Noninvolved Worker[a]		Impacts on an MEI at the Site Boundary		Impacts on Population within 50 Miles	
		Dose (rem)	Probability of an LCF[b]	Dose (rem)	Probability of an LCF[b]	Dose (person-rem)	LCFs[b,c]
Carbon Digestion in L-Area[d]							
Fire in process cell	Extremely unlikely	Because operations would be conducted in a hot cell and the facility would be provided with a sand filter, impacts are expected to be similar to H-Canyon.[e]					
Beyond-design-basis earthquake	Beyond extremely unlikely						
Melt and Dilute in L-Area							
Melter fire [f]	Extremely unlikely	31	0.04	0.16	0.0001	Not evaluated in L-Area Melt and Dilute Basis for Interim Operations (WSRC 2001a), consistent with DOE regulations.	
Full facility fire [f]	Beyond extremely unlikely	36	0.04	0.18	0.0001		
Furnace extreme overheating [f]	Beyond extremely unlikely	1.7	0.001	0.32	0.0002		
Helicopter crash [f]	Beyond extremely unlikely	<5	0.003	<0.5	0.0003		
Melter eruption with loss of ventilation [g]	Unlikely	0.71	0.0004	0.074	0.00004	3,000	2
Earthquake induced spill with loss of ventilation [g]	Beyond extremely unlikely	30	0.04	0.05	0.0003	21,000	13

LCF = latent cancer fatality; MEI = maximally exposed individual.

[a] The noninvolved worker is assumed to be 100 meters from the release point.

[b] Cancer risks were determined assuming 0.0006 LCFs per rem or person-rem and presented using one significant figure consistent with DOE guidance. For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF was doubled.

[c] The reported values are the numbers of LCFs expected to occur in the population and are presented as whole numbers; the statistically calculated values are provided in parentheses for when the reported result is 1 or less.

[d] Frequencies are on an annual basis and defined as: unlikely = 10^{-4} to 10^{-2}; extremely unlikely = 10^{-6} to 10^{-4}; beyond extremely unlikely = less than 10^{-6}.

[e] Impacts with carbon digestion in L-Area are expected to be similar to those projected for H-Canyon, with small differences due to the possible differences in filter efficiencies and distance to the MEI and nearby population. Because the dose potential for the German fuel is less than that of the "reference" spent fuel used for the accident evaluation in the *Spent Nuclear Fuel FEIS*, the impacts from the *Spent Nuclear Fuel FEIS* melter eruption and earthquake induced spill with loss of ventilation are expected to be higher than these accidents involving spent nuclear fuel from Germany.

[f] Doses reported for the noninvolved worker and MEI are based on the *L-Area Experimental Facility Basis for Interim Operation (U) (Addendum to the L-Reactor Facility BIO)* (WSRC 2001b). Because the dose potential for the German fuel is similar to the spent fuel analyzed in the accident evaluation (WSRC 2001b), the impacts from that analysis are expected to be similar to those for accidents involving the spent nuclear fuel from Germany.

[g] Doses reported for the noninvolved worker, MEI and population for the seismic spill are based on the *Spent Nuclear Fuel FEIS*, Appendix D, Table D-10 (DOE 2000b). Because the dose potential for the German fuel is less than that of the "reference" spent fuel used for the accident evaluation in the *Spent Nuclear Fuel FEIS*, the impacts from the melter eruption and earthquake induced spill with loss of ventilation are expected to be higher than those for a melter eruption or an earthquake accident involving the spent nuclear fuel from Germany

Note: To convert grams to ounces, multiply by 0.035274; miles to kilometers, by 1.6093.

Source: DOE 2000b, WSRC 2001a, 2001b.

Melt and Dilute: Under the L-Area Alternative, a melt and dilute capability would be added. The spent nuclear fuel kernels from carbon digestion would be mixed with depleted uranium and/or LEU in an alloying furnace to dilute the uranium-233 and -235 content to an acceptable concentration. Aluminum metal, potentially including SNF currently stored in L-Basin or at Idaho National Laboratory, would be added to the furnace to form an alloy with the uranium and thorium. The resulting aluminum-uranium-thorium ingots would be cooled and remotely moved from the breakout station in the furnace hot cell down a chute to a can-out capability in an adjacent lower level of the building.

Accidents associated with melt and dilute furnaces have been previously evaluated at SRS. The *FRR SNF EIS* (DOE 1996) and the *Spent Nuclear Fuel FEIS* (DOE 2000b) evaluated accidents associated with mixing melted uranium fuel with molten glass. Safety documents were prepared for a proposed, but never built, L-Area Experimental Facility (WSRC 2001a, 2001b, WSMS 2000, 2002). The facility was proposed to demonstrate a melt and dilute process for spent research reactor fuel. As with the spent nuclear fuel from Germany, the purpose was to form ingots that would be dispositioned as waste. Each of these analyses indicated that the accidents of most concern associated with melt and dilute operations were fires, energetic spills from the melter due to overpressures or steam explosions, and facility-wide events.

For purposes of radiological impact analysis, the *SRS SNF Management Final EIS* (DOE 200b) defined a “reference fuel.” The accident analysis for processing spent nuclear fuel was based on a hypothetical reference spent nuclear fuel with quantities of various isotopes selected to represent a variety of spent nuclear fuels that might be processed at SRS. This reference fuel combined the radiological characteristics of many types of SNF from both U.S. production reactors, research reactors, and various test reactors. Analyses developed by SRS using this reference fuel were intended to set a reasonable upper bound on potential radiological impacts to workers and the public from both routine operations and accidents. Using this reference fuel for analyses allowed SRS to establish health and safety controls on operations to ensure that management of SNF, including receipt, storage, processing in H-Canyon, and waste management operations could be conducted safely.

The dose-effectiveness of the mix of isotopes in the German fuel was compared to the mix of isotopes in the *Spent Nuclear Fuel FEIS* reference fuel. The German fuel that would be sent to SRS is expected to fall within the range of the radiological characteristics of the reference fuel developed by SRS and used in their safety analyses and in the radiological impacts analysis. That comparison indicated that after the German fuel has undergone carbon digestion and the fuel kernels are separated from the graphite, the radiological impacts for similar accidents should be no greater than those for accidents evaluated in the *Spent Nuclear Fuel FEIS.* Where practicable, the *Spent Nuclear Fuel FEIS* accident scenarios and impacts were incorporated into this EA.

The preliminary accident analysis for the L-Area Experimental Facility was performed using University of Missouri Materials Test Reactor fuel assemblies as the base case (WSRC 2001b). The safety analysis (basis for interim operation) for the demonstration facility indicated that the highest consequence credible (extremely unlikely) accident would be a furnace area fire occurring concurrent with melting the fuel. The resulting MEI dose was 0.16 rem. The dose-effectiveness (that is, the radiological dose from inhalation exposure to different mixes of radionuclides) of the mix of isotopes in the German fuel was compared to the mix of isotopes in the University of

Missouri Materials Test Reactor fuel evaluated in the L-Area Experimental Facility accident analysis (WSRC 2001b).

Based on a comparison of the inhalation doses that would be received from the radionuclides in the fuels, radiological impacts of melt and dilute accidents with German fuel are expected to result in similar consequences as those from University of Missouri Materials Test Reactor fuel when similar quantities of fuel are processed. Operational accidents judged "beyond extremely unlikely" included a full facility fire concurrent with melting the fuel and a furnace extreme overheating event; the corresponding MEI doses were 0.18 and 0.32 rem, respectively. External events including a helicopter crash resulted in an MEI dose of less than 0.5 rem. Similar accidents and impacts would be expected for German fuel with similar operations and MAR. The University of Missouri spent fuel was low burnup (150 MW-day), but with only 150 days of cooling. The effective dose for German fuel for these melt and dilute accidents is estimated to be similar to that for the University of Missouri fuel.

Table 4-20 presents the estimated doses for L-Area accidents. Accidents scenarios for carbon digestion in L-Area are assumed to be the same as those projected for H-Area, with similar materials at risk, releases, and impacts. Accidents for the melt and dilute process for the spent nuclear fuel from Germany in L-Area are based on the accident analyses for the L-Area Experimental Facility (WSRC 2001b) and for the *Spent Nuclear Fuel FEIS* (DOE 2000b).

4.1.3.2.2.7 Waste Disposition Accidents under the H-Area and L-Area Alternatives

Under all processing options for the H-Area Alternative, no major changes would be expected in operations at the existing HLW tank farm; the HLW pretreatment infrastructure; DWPF and the glass waste storage facilities in S-Area; the saltstone facilities in Z-Area; and the waste management capabilities at E-Area. Therefore, no substantial changes in accident risks from continued operation of these facilities would be expected among members of the public or workers. However, under all three options under the H-Area Alternative the additional HLW that would be generated would extend the operating periods of these facilities. Consequently, members of the public would be exposed to potential accident risks from these facilities for that much longer. Under the Vitrification Option, operation of DWPF would be extended by approximately 100 days while operation of the saltstone facilities and the future Salt Waste Processing Facility would be extended by approximately 24 days. Under the LEU Waste and LEU/Thorium Waste Options, operation of DWPF, the future Salt Waste Processing Facility, and the saltstone facilities would each be extended by approximately 30 days (DOE 2014a).

Under the L-Area Alternative, low-activity radioactive liquids would be generated that would be processed into grout at the Saltstone Production Facility. Operation of the saltstone facilities under this Alternative would be extended by approximately 16 days.

Accidents and potential impacts on the public from operation of DWPF and the saltstone facilities have been addressed in previous NEPA documentation (DOE 1994b, 2001a). The potential radiological impacts on the noninvolved worker, MEI, and offsite population based on these previous analyses are presented in **Table 4-21**.

4.1.3.2.3 Intentional Destructive Acts

At SRS, the spent nuclear fuel from Germany would be protected and processed such that an intentional destructive act that would threaten the public or workers would be extremely unlikely. The fuels would be stored in heavily shielded casks in a property protection area while awaiting processing. For most of the process, the fuel would be within the hot cells of heavily reinforced buildings. Intentional destructive acts at the SRS H-Canyon have been previously analyzed in a classified appendix of the *Final Surplus Plutonium Disposition Supplemental Environmental Impact Statement* (*Final SPD SEIS*) (DOE 2015a). That analysis evaluated potential impacts in the terms of consequences (that is, impacts if the event were to occur). The impacts of the highest consequence intentional destructive act in that analysis would be greater than the impacts of a potential event involving the spent nuclear fuel from Germany under either the H- or L-Area Alternative because the potential inhalation dose effectiveness of the German fuel is less than that of the SRS "reference fuel" used in the SRS spent nuclear fuel evaluations and much less than the dose effectiveness of the plutonium materials at risk in the SPD SEIS.

Table 4-21: Potential Accident Impacts at SRS Waste Management Facilities

Accident [a]	Frequency (per year)	Impacts on Noninvolved Worker		Impacts on an MEI at the Site Boundary		Impacts on Population within 50 Miles	
		Dose (rem)	Probability of an LCF [b]	Dose (rem)	Probability of an LCF [b]	Dose (person-rem)	LCFs [b, c]
Saltstone Activities (DOE 2001a)							
Fire in Process Cell	1.0×10^{-4}	0.14	8×10^{-5}	0.0094	6×10^{-6}	500	0 (0.30)
Beyond Design Basis Earthquake	5.0×10^{-4}	3.6	0.002	0.12	7×10^{-5}	6,100	3.7
Aircraft Impact	3.7×10^{-7}	64	0.08	2	1×10^{-3}	110,000	66
DWPF (DOE 1994b)							
Melter Spill	Extremely unlikely [d]	0.29	2×10^{-4}	0.03	2×10^{-5}	490	0 (0.29)
0.2 g Earthquake	Extremely unlikely [d]	4,000	Fatality	6.8	4×10^{-3}	76,000	46

DWPF = Defense Waste Processing Facility; LCF = latent cancer fatality; MEI = maximally exposed individual.

[a] The doses for waste management activities supporting disposition of spent nuclear fuel from Germanys are taken from EISs supporting the specified facilities.

[b] Cancer risks were determined assuming 0.0006 LCFs per rem or person-rem and presented using one significant figure consistent with DOE guidance. For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF was doubled; doses of 600 rem or more are assumed to result in a near-term fatality.

[c] The reported values are the numbers of LCFs expected to occur in the population and are presented as whole numbers; the statistically calculated values are provided in parentheses when the reported result is 1 or less.

[d] Frequencies are on an annual basis and defined as: extremely unlikely = 10-6 to 10-4; beyond extremely unlikely = less than 10-6.

Note: To convert grams to ounces, multiply by 0.035274; miles to kilometers, by 1.6093.

Source: DOE 1994b and 2001a.

4.1.3.2.4 Chemical Environment

The inventories for most chemicals at SRS facilities are small, and because of SRS's remote location and large size, there is minimal risk of chemical exposure to the surrounding population resulting from normal site operations or accidents. Nevertheless, chemical release monitoring is regularly performed.

The potential for hazardous chemical impacts on noninvolved workers and the public has been evaluated for many of the facilities that might use or store larger quantities of hazardous chemicals (SRNS 2010; WGI 2005), no substantial impacts from operations or accidents were found for noninvolved workers or the public.

None of the alternatives are expected to substantially change the chemical exposures at SRS. There are minor introductions of chemicals and industrial gases to either H-Canyon or L-Area during the construction phase under any of the alternatives. These have been identified as approximately 100 gallons of construction-related chemicals, and from 20 to 800 cubic meters of industrial gases primarily used for welding (such as acetylene, oxygen, carbon dioxide/argon, or helium) (DOE 2014a). Very small quantities of solid and liquid hazardous wastes may be generated during construction.

No chemicals are expected to be used during the receipt, storage or transfer of CASTOR casks at SRS. Accordingly, no chemical exposures are expected during those activities.

Chemicals used during processing in H-Area and L-Area would be managed in accordance with established procedures for safe handling. No unusual or unique hazardous chemicals would be needed or generated under the alternatives being considered. **Table 4-22** shows the potential chemical use for operations under both action alternatives. As noted in the discussion of radiological risk and safety, Savannah River Nuclear Solutions prepared a *Safety in Design Tailoring Strategy for HTGR Fuel Receipt and Disposition Feasibility Study,* (SRNS 2014d) to describe the overall safety approach to be taken for the German fuel acceptance and disposition.

Table 4-22: Annual Chemical Use for Operations

Chemical	H-Area Alternative			L-Area Alternative
	Option 1 - Vitrification	Option 2 – LEU Waste	Option 3 – LEU/Thorium Waste	Melt and Dilute
Aluminum (kg/yr)	N/A	N/A	N/A	13,000
Aluminum nitrate (kg/yr)	5,200	5,200	5,200	N/A
Argon (l/yr)	0	500,000	500,000	0
Boric Acid (kg/yr)	77	23	23	0
Calcium or Magnesium (kg/yr)	N/A	N/A	N/A	2,800
Copper formate (kg/yr)	650	200	200	N/A
Fly ash (kg/yr)	0	10,000	10,000	0
Formic Acid (kg/yr)	25,000	7,600	7,600	N/A
Glass frit (kg/yr)	260,000	78,000	78,000	0
Hydrogen peroxide (kg/yr)	8,000	8,000	8,000	8,000
Nitric acid (kg/yr)	330,000	290,000	290,000	210,000
Nitrogen (l/yr)	0	2,000	2,000	0
Oxalic Acid (kg/yr)	66,000	20,000	20,000	58
Portland cement (kg/yr)	0	30,000	30,000	0
Potassium fluoride (kg/yr)	500	500	500	N/A
Potassium nitrate (kg/yr)	77	23	23	0
Saltstone Premix (kg/yr)	4,800,000,000	4,800,000,000	4,800,000,000	2,600,000,000
Slag (kg/yr)	14,000,000	4,100,000	4,100,000	0
Sodium hydroxide (kg/yr)	680,000	280,000	280,000	52,000
Sodium nitrate (kg/yr)	190,000	140,000	140,000	120,000
Sodium tetraphenylborate (kg/yr)	5,800	1,700	1,700	0
Stainless steel 304L (kg/yr)	26,000	46,000	46,000	14,000
Sodium titanate (kg/yr)	94,000	28,000	28,000	N/A
Uranium, depleted (metric tons)	0	3.2	3.2	3.2
Zeolite, monosodium titanate, crystalline silicotitanate (kg/yr)	100	190	190	300
Zirconium oxide (kg/yr)	N/A	10,000	10,000	N/A

N/A – not applicable

Note: to convert kilograms to pounds, multiply by 2.2

Source: Estimates of projected chemical usage during operation were developed based on DOE projections provided in the following documents: the *Final Supplemental Environmental Impact Statement, Defense Waste Processing Facility* (DOE 1994); the *Savannah River Site, Salt Processing Alternatives Final Supplemental Environmental Impact Statement* (DOE 2001a), and DOE 2014a.

A consolidated hazards analysis process would be used to identify accident scenarios, assess consequences, and guide development of controls (DOE 2014a). H-Canyon processes have previously been evaluated pursuant to safety class and safety significant systems, structures and components (SSCs). These SSCs include controls to prevent or mitigate the consequences of accidents, to well below evaluation guidelines. SSCs include the building structure, canyon exhaust ventilation system, sand filter, backup diesel generators, various monitoring, alarm and interlock systems, and vessel air purge system (DOE 2014a). Process-specific SSCs controls would be identified to prevent the exposure of a worker to a concentration of hazardous material in an occupied area inside a building, as determined by uniform distribution of the released material in the occupied area that would challenge a concentration of Protective Action Criteria (PAC)-3 (DOE 2014a). Safety Significant controls would also be required to ensure that any credible event shall not exceed the threshold value of PAC-3 for a Collocated Worker Chemical Evaluation Criteria or a PAC-2 to an individual member of the public based on the analysis approach described in DOE-STD-1189 (DOE 2014a). This same process would be applied to evaluate necessary process safety controls for activities in L Area.

Accordingly, accidental human health exposure to any of the chemicals proposed for use in the proposed action is deemed unlikely.

4.1.3.3 Socioeconomics

As described in Section 3.3.3, the socioeconomic ROI for SRS is defined as the four-county area of Columbia and Richmond counties in Georgia, and Aiken and Barnwell counties in South Carolina. Potential impacts from construction and operations are discussed separately, although there may be some overlap in construction of the uranium solidification facility and operations occurring under the H-Area Alternative (LEU and LEU/Thorium Waste Options).

Construction - Under the H-Area Alternative, construction would require up to 100 employees under the Virtification Option and up to 201 employees under the LEU and LEU/Thorium Waste Options. Up to 155 employees would be required under the L-Area Alternative (DOE 2014a). Both the H-Area and L-Area Alternatives would generate mostly new construction jobs (approximately 1 to 3 percent of the 7,224 persons employed at SRS). No modifications are expected for the DWPF or the saltstone facilities, therefore no additional construction jobs would be created for these facilities. Although both alternatives would result in some job creation, the numbers of jobs and the duration of employment are not expected to result in a noticeable impact to the existing socioeconomic or demographic characteristics of the region.

Operations - Peak employment during operations under either alternative would be 125 to 150 persons. The Vitrification Option would not create additional jobs. Under the LEU Waste and LEU/Thorium Waste Option as many as 20 new or reassigned SRS employees (less than 1 percent of the 7,224 persons employed at SRS) would be required for operation of the uranium solidification facility, which would not likely result in a noticeable change in existing socioeconomic or demographic characteristics. Under the H-Area Alternative, no major changes would be expected in operations at the existing DWPF, saltstone facilities, or the future Salt Waste Processing Facility. Therefore, no additional employees would be needed at these facilities and there would be no socioeconomic impacts. However, under all three options the additional radioactive waste generated would extend the operating periods of these facilities thereby preserving existing jobs (including direct and indirect jobs). Under the Vitrification Option, operation of DWPF would be extended by approximately 100 days while operation of the saltstone facilities and the future Salt Waste Processing Facility would be extended by approximately 24 days. Under the LEU Waste and LEU/Thorium Waste Options, operation of DWPF, the future Salt Waste Processing Facility, and the saltstone facilities would each be extended by approximately 30 days (DOE 2014a).

The L-Area Alternative would preserve approximately 135 jobs. As a result, there would be a small beneficial impact in the ROI by preserving existing jobs at SRS. Under the L-Area

Alternative, operation of the saltstone facilities would be extended by approximately 16 days (DOE 2014a).

4.1.3.4 Waste Management

The types and volumes of wastes that would be generated by the activities evaluated in this EA are summarized in **Table 4-23**. Table 4-23 also indicates parenthetically the percent of SRS waste management capacity for that waste type, as compared to the existing SRS waste management capacities summarized in **Table 4-24**. The projected volumes would be within waste management capacities at SRS. However, as discussed for solid LLW in this section, additional analysis and facility design or operational modifications may be required to accommodate disposal of solidified LEU or LEU/Thorium waste under the H-Area Alternative.

Table 4-23: Waste Generation and Percent of SRS Waste Management Facility Capacity

Waste Type	H-Area Alternative[a]			L-Area Alternative[a]
	Vitrification Option	LEU Waste Option	LEU/Thorium Waste Option	
Construction				
Solid LLW (cubic meters)	320 (0.1)	320 (0.1)	320 (0.1)	390 (0.1)[b]
Solid hazardous (cubic meters)	0.15 (0.02)	1.7 (0.3)	1.7 (0.3)	NG
Liquid hazardous (liters)	190 (0.02)	570 (0.1)	570 (0.1)	NG
Solid nonhazardous (cubic meters)	110 (0.0009)	340 (0.004)	340 (0.004)	NG
Liquid nonhazardous (liters)	9,500 (0.0002)	32,000 (0.001)	32,000 (0.001)	NG
Operations				
Solid LLW (cubic meters)	2,000 (0.7)	2,300 (0.8)	2,500 to 2,900 (0.9 to 1.0)[c]	2,000 (0.7)
Liquid LLW (liters)	NG	280,000 (0.03)	280,000 (0.03)	NG
Hazardous (cubic meters)	NG	0.15 (0.03)	0.15 (0.03)	NG
Solid nonhazardous (cubic meters)	NG	75 (0.001)	75 (0.001)	NG
Liquid nonhazardous (liters)	NG	2,800,000 (0.1)	2,800,000 (0.1)	NG
HLW canisters or MCOs (number)	101 (2)	32 (0.7)	15 (0.3)	82 (NA)[d]
Saltstone grout (liters)[e]	5,500,000 (16-24)	6,200,000 (18-27)	6,200,000 (18-27)	3,700,000 (5-8)

HLW = high-level radioactive waste; LEU = low-enriched uranium; LLW = low-level radioactive waste; MCO = multi-canister overpack; NA = not applicable; NG = not generated in meaningful quantities.

[a] The values in parentheses represent the percent of existing SRS waste management capacity represented by the projected volume for that waste type, using the waste management capacities listed in Table 4-24.

[b] The LLW volume includes about 15 cubic meters of LLW in demolition debris that may also contain polychlorinated biphenyls in paint. This waste may be disposed of as LLW in E-Area (DOE 2014e).

[c] The lower value includes about 155 cubic meters of solidified LEU/thorium waste that may need additional packaging to provide shielding to reduce the radiation dose from the package. If so, the total volume of LLW under this option could increase to about 2,900 cubic meters, representing about 1 percent of the E-Area volume capacity.

[d] MCOs from melt and dilute operations at L-Area would be stored in concrete storage overpacks on an L-Area pad rather than at S-Area storage locations. Sufficient storage capacity would be available.

[e] The quantity of saltstone grout is the total for the project duration (approximately 3.5 years for the H-Area Alternative and 7 years for the L-Area Alternative); however, the percent of capacity (value in parenthesis) is based on the annual saltstone processing rate.

Note: Calculated values have been rounded. To convert cubic meters to cubic feet, multiply by 35.314; liters to gallons, multiply by 0.26418.

Source: DOE 2014a.

Table 4-24: Summary of Existing Waste Management Capacities at the Savannah River Site

Waste Type	Capacity	Disposition Method
HLW Canisters from DWPF	4,590 canisters	Onsite storage in S-Area
Solid LLW	276,300 cubic meters [a]	Onsite disposal vaults, slit trenches, or engineered trenches
Liquid LLW	590,000,000 liters per year	Onsite F/H Effluent Treatment Project
Saltstone	6,700,000 to 10,000,000 liters per year [b]	Onsite Saltstone Production Facility
Hazardous	296 cubic meters [c]	Onsite storage pads
Solid nonhazardous	4,200,000 cubic meters per year	Regional municipal waste landfill disposal
Liquid nonhazardous	1,500,000,000 liters per year	Onsite Central Sanitary Wastewater Treatment Facility

DWPF = Defense Waste Processing Facility; HLW = high-level radioactive waste; LLW = low-level radioactive waste.

[a] As of October 2014, the estimated unused disposal capacity remaining is approximately 21,300 cubic meters for the Low-Activity Low-Level Radioactive Waste Vaults, 180,000 cubic meters for the slit trenches, and 75,000 cubic meters for the engineered trenches. See Chapter 3, Section 3.3.4.1.

[b] Current approximate annual production of saltstone at the Saltstone Production Facility (SRR 2105), assuming about 1.76 gallons of saltstone per gallon of solution input to the facility (SRR 2014b). When the Salt Waste Processing Facility is operational, the production rate is projected to increase to about 40,000,000 liters per year (SRR 2015).

[c] E-Area storage pads are permitted to store up to 296 cubic meters of hazardous waste or mixed low-level radioactive waste.

Note to convert cubic meters to cubic feet, multiply by 35.314; liters to gallons, multiply by 0.26418.

Source: DOE, 2013c; 2015; Maxted 2014; SRR 2014b, 2015.

Solid Low-Level Radioactive Waste. Under the H-Area Alternative, solid LLW would be generated from modifications to H-Canyon to install a carbon digestion capability, from operations at H-Canyon to remove spent nuclear fuel from CASTOR cask liners (metal canisters) and from operations at the uranium solidification facility under the LEU Waste and LEU/Thorium Waste Options.

Solid LLW from modifications to H-Canyon to install a carbon digestion capability is expected to consist primarily of surface-contaminated metals including decommissioned process tanks and piping. The contaminated metal waste would be placed in boxes for transport to E-Area for disposal, offsite at NNSS, or offsite at a commercial disposal facility. The projected 320 cubic meters (11,300 cubic feet) of LLW would represent approximately 0.1 percent of the SRS LLW disposal capacity, and would not impact the SRS waste management infrastructure.

Operations under the Vitrification Option would generate LLW consisting primarily of the CASTOR casks and metal canisters from the CASTOR casks, which would not be used for disposal of the waste form generated under this option. These casks and canisters could be internally activated or contaminated with residual radioactive materials from the spent nuclear fuel. Small quantities of job control LLW (e.g., personal protective equipment, filters, and empty containers) could be also generated. The CASTOR casks would be used as disposal containers for the empty metal canisters, which would be replaced in the casks, the cask lids would be reinstalled, and the casks would be transferred to E-Area, offsite to NNSS, or offsite to a commercial disposal facility. Each cask transferred for disposal would weigh approximately 30 metric tons (33 tons). It is expected that casks of this weight could be handled at E-Area or at offsite disposal facilities (DOE 2014a).

The total LLW volume is projected to be about 2,000 cubic meters (72,000 cubic feet), which would represent 0.7 percent of the SRS LLW disposal capacity. The casks would comprise about 1,900 cubic meters (67,000 cubic feet) of this LLW volume (determined based on the outer dimensions of the casks) and the remaining volume, about 140 cubic meters (4,900 cubic feet), would consist of job control LLW. It is assumed that this job control LLW would be packaged in drums or boxes for disposal at E-Area.

Operations under the LEU Waste and LEU/Thorium Waste Options would generate LLW consisting primarily of empty metal canisters from the CASTOR casks, solidified LEU or LEU/thorium from the uranium solidification facility, and additional LLW from operation of the uranium solidification facility. Under both options the empty metal canisters would be packaged to preclude the release of contamination during handling, transfer, and disposal, and disposed in E-Area as LLW. The solidified LEU or LEU/thorium waste would be disposed at E-Area, offsite at NNSS, or offsite at a commercial disposal facility.

Under the LEU Waste Option, the operational LLW would total about 2,300 cubic meters (81,000 cubic feet), which would represent about 0.8 percent of the SRS LLW disposal capacity. Most of this waste, approximately 1,900 cubic meters (67,000 cubic feet) as determined by the outer dimensions of the casks, would consist of the solidified LEU waste form in metal containers inside sealed CASTOR casks.

Additional LLW generated under this option would include about 220 cubic meters (7,900 cubic feet) of empty metal canisters (CASTOR cask liners), about 140 cubic meters (4,900 cubic feet) of job control waste, and about 75 cubic meters (2,600 cubic feet) of LLW from operation of the uranium solidification facility. This waste would be disposed of in E-Area.

Under the LEU/Thorium Waste Option, the operational LLW would range from 2,500 to 2,900 cubic meters (88,000 to 100,000 cubic feet), which would represent about 0.9 to 1.0 percent of the SRS LLW disposal capacity. Solidification of LEU/thorium waste under this option would generate more containers of solidified waste than would fit into the empty CASTOR casks. The remaining containers, comprising about 155 cubic meters (5,500 cubic feet) of solidified LEU/thorium waste (in addition to the 1,900 cubic meters [67,000 cubic feet] contained in the CASTOR casks), could be disposed in their as-solidified form, or placed into overpacks (such as concrete culvert sections) to provide shielding, if storage is required before disposal or to meet disposal facility requirements. Assuming concrete culvert sections are used to provide shielding, it is estimated that the volume of the waste form would increase by about a factor of 2.4. In this case, the volume of the remaining solidified LEU/thorium waste would range from 155 cubic meters (5,500 cubic feet) to 365 cubic meters (12,900 cubic feet).

Additional LLW generated under this option would include about 220 cubic meters (7,900 cubic feet) of empty metal canisters (CASTOR cask liners), about 140 cubic meters (4,900 cubic feet) of job control waste, and about 75 cubic meters (2,600 cubic feet) of LLW from operation of the uranium solidification facility. This waste would be disposed of at E-Area.

Casks with containers of solidified LEU or LEU/thorium waste may require temporary storage in H-Area or E-Area pending disposition. Ingrowth of uranium-232 progeny would occur during storage, resulting in an increase in radiation levels at the surfaces of the containers holding the solidified LEU or LEU/thorium. Under the LEU Waste Option, the solidified LEU would be inside

CASTOR casks, which would provide shielding during storage. Under the LEU/Thorium Waste Option, the additional solidified LEU/thorium waste containers not in CASTOR casks could be placed in concrete culvert sections (or other secondary containment) to provide additional shielding during storage and/or handling.

The total quantity of fissile isotopes expected to be present in the solidified LEU and LEU/thorium waste may not meet current waste acceptance criteria at the E-Area disposal facility, offsite at NNSS, or offsite at a commercial disposal facilities. In accordance with DOE's *Radioactive Waste Management Manual*, DOE M 435.1-1 (DOE 2001b), wastes are required to have an identified path to disposal prior to generation. Therefore, DOE would need to have a reasonable expectation that the solidified LEU or LEU/thorium waste could be disposed at an authorized DOE or commercial facility prior to its generation, and would need to have plans and activities in place for achieving final disposal of the waste. Reviews of waste acceptance criteria and performance assessments for E-Area, NNSS, or commercial disposal facilities would be performed to determine whether the waste could be disposed under existing facility configurations and operating procedures or whether either or both would require modification. These reviews would be performed in accordance with requirements in Section IV(P)(7) of DOE's *Radioactive Waste Management Manual* (DOE 2001b) and Section 5.1.13 of the *SRS Radioactive Waste Requirements* manual (DOE 2014e) that allow for consideration of waste streams that have not been previously identified for disposal. These reviews would determine whether disposal of the waste would be in accordance with the performance objectives in DOE's *Radioactive Waste Management Manual* (DOE 2001b), as well as the safety requirements for LLW disposal, including criticality safety. It might be that use of less conservative parameter values in the performance assessment (for example, assuming the casks provide some reduction in releases to the environment) would show that the material can be safely disposed of with no changes in disposal practices; or additional engineering features such as emplacement of the waste within a specially designed disposal unit featuring additional barriers against long-term releases to the environment or inadvertent human intrusion could be required. These barriers could include, for example, a grout backfill (SRNL 2014d).

Under the L-Area Alternative, solid LLW could be generated from modifications to L-Area to install carbon digestion and melt and dilute capabilities including installing an air treatment system, and from operations to remove the spent nuclear fuel from Germany from the metal cask liners (canisters) for feed to the carbon digestion capability. During construction, approximately 390 cubic meters (13,800 cubic feet) of LLW would be generated over 4 years. This LLW would include about 15 cubic meters (530 cubic feet) of waste that may also contain polychlorinated biphenyls associated with paint that was used when the facility was built. It is expected that this waste would be acceptable for disposal at E-Area consistent with the E-Area waste acceptance criteria (DOE 2014e). The 390 cubic meters (13,800 cubic feet) of LLW would represent 0.1 percent of the SRS LLW disposal capacity.

During operations, the total LLW volume is projected to be about 2,000 cubic meters (72,000 cubic feet). The empty CASTOR casks under the L-Area Alternative would be used as disposal containers for the metal cask liners, and would comprise about 1,900 cubic meters (67,000 cubic feet) of the total LLW volume. The remaining volume, about 140 cubic meters, (4,900 cubic feet) would consist of job control LLW. The 2,000 cubic meters (72,000 cubic feet) of LLW would represent about 0.7 percent of the SRS LLW disposal capacity.

Liquid Low-Level Radioactive Waste. Under the H-Area Alternative, LEU Waste and LEU/Thorium Waste Options, liquid LLW would be generated during operation of the uranium solidification capability. Liquid LLW would be piped from H-Area to the Effluent Treatment Project for treatment to remove radionuclides; the treated effluent would be discharged in compliance with regulatory requirements through an NPDES-permitted outfall to Upper Three Runs. Because the total quantity of liquid LLW projected from the uranium solidification facility (280,000 liters [75,000 gallons]) represents only 0.03 percent of the annual capacity of the Effluent Treatment Project, there would be no impact on SRS waste management capacity for this waste.

Hazardous waste. About 0.15 cubic meters (5.3 cubic feet) of solid hazardous waste and 190 liters (50 gallons) of liquid hazardous waste would be generated from installation of a carbon digestion capability at H-Canyon under each of the three H-Area Alternative processing operations. This would be the total amount of hazardous waste generated under the Vitrification Option.

Construction of a uranium solidification capability under the LEU Waste and LEU/Thorium Waste Options would generate another 1.5 cubic meters (53 cubic feet) of solid hazardous waste and about 380 liters (100 gallons) of liquid hazardous waste. Considering installation of both carbon digestion and uranium solidification capabilities, construction under the LEU Waste and LEU/Thorium Waste Options would generate a total of 1.7 cubic meters (60 cubic feet) of solid hazardous waste and 570 liters (150 gallons) of liquid hazardous waste. Hazardous waste from construction would be temporarily stored, as needed, on onsite storage pads and transported offsite for treatment and disposal. Hazardous waste generation under any processing option would represent less than 1 percent of the SRS storage capacity for this waste and would not impact the SRS waste management infrastructure. No meaningful quantities of hazardous waste are expected from construction under the L-Area Alternative.

During operations, hazardous waste in an appreciable quantity would be generated only under the H-Area Alternative, LEU Waste and LEU/Thorium Waste Options. A total of 0.15 cubic meters (5.3 cubic feet) of solid hazardous waste would be generated that would be shipped offsite for treatment and disposal. There would be no impact to SRS waste management capacity for this waste; even if all waste was generated in a single year and required temporary storage pending offsite shipment, 0.15 cubic meters (5.3 cubic feet) of waste would represent only 0.03 percent of the SRS storage capacity for this waste. This volume of waste would have no impact to SRS waste management infrastructure.

Nonhazardous waste. Under the H-Area Alternative, solid and liquid nonhazardous wastes would be generated under all processing options from installation of a carbon digestion capability at H-Canyon. Installation of a carbon digestion capability would generate about 110 cubic meters (3,900 cubic feet) of solid nonhazardous waste and 9,500 liters (2,500 gallons) of liquid nonhazardous waste. This would be the only nonhazardous waste generated during construction under the Vitrification Option.

Construction of a uranium solidification capability under the LEU Waste and LEU/Thorium Waste Options would generate another 22,700 liters (6,000 gallons) of liquid nonhazardous waste. Considering installation of both carbon digestion and uranium solidification capabilities, construction under the LEU Waste and LEU/Thorium Waste Options would generate a total of about 340 cubic meters (12,000 cubic feet) of solid nonhazardous waste and 32,000 liters (8,500 gallons) of liquid nonhazardous waste. No meaningful quantities of nonhazardous waste are expected from construction under the L-Area Alternative. Under both the H-Area and L-Area Alternatives, small quantities of solid nonhazardous waste could be generated as part of construction of a cask storage capability in L- and/or H-Area.

Solid nonhazardous waste would be disposed of in the onsite Three Rivers Regional Landfill or an onsite construction and demolition landfill. Liquid nonhazardous waste would be piped to the Central Sanitary Wastewater Treatment Facility for treatment before discharge in compliance with regulatory requirements to an NPDES-permitted outfall. Under any alternative or processing option, solid and liquid nonhazardous waste generation would represent 0.001 percent or less of the SRS disposal or treatment capacity. No impacts would be expected on the SRS waste management infrastructure.

During operation of the uranium solidification facility under the H-Area Alternative, LEU Waste and LEU/Thorium Waste Options, nonhazardous solid and liquid wastes would be generated. These wastes would represent about 0.001 percent and 0.1 percent of the annual capacities at the Three Rivers Regional Landfill and Central Sanitary Wastewater Treatment Facility, respectively. No impacts would be expected on the SRS waste management infrastructure.

High Level Radioactive Waste. Under the three H-Area Alternative processing options, liquid HLW generated at H-Canyon would be stored in the SRS tank farm system, pretreated, and vitrified at DWPF. HLW canisters from DWPF would be transferred to the glass waste storage facilities in S-Area. Under the Vitrification Option, 101 canisters of vitrified HLW would be generated; 32 canisters would be generated under the LEU Waste Option, and 15 canisters would be generated under the LEU/Thorium Waste Option. As discussed in Chapter 3, Section 3.3.4.2, as of December 31, 2013, 3,754 canisters had been poured at DWPF; the estimated storage capacity at the existing two Glass Waste Storage Buildings is about 4,590 canisters and DOE has plans to provide additional storage capacity by December 2018 (SRR 2014b). Under all H-Area Alternative options, it is expected that there would be sufficient capacity at S-Area to safely store all canisters of vitrified HLW pending disposition.

Under the L-Area Alternative, the melt and dilute process would generate aluminum-uranium-thorium ingots that would be placed into multi-canister overpacks (MCOs) that would each hold 28 ingots in two layers, each layer comprising a basket of 14 ingots. The MCOs would be loaded into concrete overpacks for storage. Up to five MCOs, each 2 feet in diameter and almost 14 feet long, would be stored in each concrete overpack. This alternative is projected to generate 82 MCOs. It is expected that there would be sufficient storage capacity in L-Area to safely store all MCOs pending disposition.

Saltstone. Saltstone grout would be generated under the three H-Area Alternative process options and the L-Area Alternative. The Vitrification Option would result in disposal of approximately 5.5 million liters (1.45 million gallons) of saltstone grout, representing about 16 percent to 24 percent of the current annual production capability of the Saltstone Production Facility, which is about 6.7 million to 10 million liters (1.76 million gallons to 2.64 million gallons) (see Table 4-24). The LEU Waste and LEU/Thorium Waste Options would result in disposal of approximately 6.2 million liters (1.65 million gallons) of saltstone grout, or about 18 percent to 27 percent of the current annual production capability of the Saltstone Production Facility. The H-Area Alternative volumes would represent 0.8 to 0.9 percent of the 662 million liters (175 million gallons) of waste expected to require treatment and disposal at the Saltstone facilities (DOE 2001a).

Because the projected saltstone volumes are small compared to the current production capability of Saltstone Production Facility and the expected increased capability when the Salt Waste Processing Facility becomes operational (SRR 2015), the volumes of grout projected for the proposed action would be accommodated within the SRS grout production and disposal program. Therefore, no impacts would be expected on the SRS waste management infrastructure.

Under the L-Area Alternative, about 3.7 million liters (970,000 gallons) of saltstone grout would be generated, representing about 5 percent to 8 percent of the current annual production capability of the Saltstone Production Facility. The L-Area Alternative volume would represent approximately 0.6 percent of the 662 million liters (175 million gallons) of waste expected to require treatment and disposal at the Saltstone facilities (DOE 2001a). As with the H-Area Alternative, no impacts would be expected on the SRS waste management infrastructure.

4.1.3.5 Transportation

4.1.3.5.1 Methodology

This section presents the potential transportation risks associated with incident-free and accident conditions for each of the action alternatives and options described in Chapter 2. Transportation of spent nuclear fuel from Joint Base Charleston – Weapons Station to SRS would be required under each alternative. Transportation of radioactive waste from SRS to either NNSS or an offsite commercial disposal facility might occur under options involving disposal of LEU or LEU/thorium as a grouted LLW. Transportation accidents involving radioactive materials have the potential for both radiological and nonradiological risk to transportation workers and the public.

In determining transportation risks, per-shipment risk factors were calculated for incident-free and accident conditions using the Radioactive Material Transportation Risk Assessment Code Version 6.02 (RADTRAN 6.02) computer program (SNL 2013), in conjunction with the Transportation Routing Analysis Geographic Information System (TRAGIS) computer program (Johnson and Michelhaugh 2003). RADTRAN 6.02 was used to estimate the impacts on transportation workers and members of the public. For incident-free transportation, the potential human health impacts of the radiation fields surrounding the transportation packages were estimated for transportation workers and the population along the route (people living along the route), as well as for people sharing the route (car occupants along the route) and at rest areas and other stops along the route. For incident-free and accident conditions, the affected population included individuals living within 0.5 miles (0.8 kilometers) and 50 miles (80 kilometers) of each side of the road or railroad, respectively.

Radiological health impacts are expressed in terms of additional LCFs. LCFs associated with radiological exposure are estimated by multiplying the occupational (worker) and public dose by a dose conversion factor of 0.0006 LCFs per rem or person-rem of exposure (DOE 2003). Nonradiological accident impacts are expressed as additional immediate (traffic accident) fatalities. The assumptions and resulting risk estimates are presented in the following sections.

4.1.3.5.2 Offsite Route Characteristics

Route characteristics that are important to the radiological risk assessment include the total shipment distance and population distribution along the route. TRAGIS was used to map transportation routes in accordance with U.S. Department of Transportation (DOT) regulations. The TRAGIS program provides population density estimates for rural, suburban, and urban areas along transportation routes based on 2010 census data and the distance traveled in each area (see **Table 4-25**). Route-specific accident and fatality rates for commercial truck and rail transportation were used to determine the risk of traffic accident fatalities (Saricks and Tompkins 1999) after adjusting for possible under-reporting in truck rates (UMTRI 2003).

Table 4-25: Offsite Truck and Rail Route Characteristics

Origin	Destination	Method	Nominal Distance (miles)	Distance Traveled in Zones (miles)			Population Density in Zone[a] (number per square mile)			Number of Affected Persons[b]
				Rural	Suburban	Urban	Rural	Suburban	Urban	
Joint Base Charleston	SRS	Rail	133	80	48	5	32	867	9,468	88,588
SRS	Commercial disposal facility [c]	Rail	2,522	1,514	824	184	36	1,161	11,711	3,166,880
AREVA[d]	SRS	Rail	2,977	1,846	893	238	37	1,048	10,353	3,471,041
SRS	NNSS	Truck	2,213	1,479	673	61	48	825	7,829	1,103,463
SRS	Commercial disposal facility [c]	Truck	2,026	1,225	711	90	37	923	8,323	1,453,060
AREVA[d]	SRS	Truck	2,540	1,592	863	85	42	922	8,358	1,571,089
SRS	Intermodal terminal [e]	Rail	2,784	1,823	802	159	29	1,104	10,740	2,648,878
Intermodal terminal	NNSS[e]	Truck	166	156	10	0	10	803	0	9,686

CA = California; DOE = U.S. Department of Energy; LLW = low-level waste; NNSS = Nevada National Security Site; SRS = Savannah River Site; WA = Washington

[a] Population densities have been projected to 2020 using state-level data from the 2010 census (Census 2011a) and assuming state population growth rates from 2000 to 2010 continue to 2020.

[b] The estimated number of persons residing within 0.5 miles along the transportation route, projected to 2020.

[c] In order to generate conservative results, it was assumed that if LEU or LEU/thorium waste were shipped to a commercial disposal facility (either Energy Solutions in Utah or Waste Control Specialists in Texas), it would be shipped to the site that would result in the larger impacts.

[d] Depleted uranium would be used in the down blending process. To be conservative, a shipment of depleted uranium (uranyl nitrate [liquid form]) (DUNH) from AREVA (located in Richland, WA) to SRS was analyzed. AREVA was used because it provided the most conservative rail and truck route characteristics.

[e] Intermodal transportation may be required to transport CASTOR casks to NNSS. In this case, the CASTOR casks would be transported to a nearby location (intermodal terminal) via rail and then transported by truck to the DOE LLW disposal facility.

Note: To convert from miles to kilometers, multiply by 1.60934; to convert from number per square mile to number per square kilometer, multiply by 0.3861. Rounded to the nearest mile.

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4.1.3.5.3 Radioactive Material Shipments

Shipping packages containing radioactive materials emit low levels of radiation; the amount of radiation depends on the kind and amount of transported materials and the packaging. DOT regulations (49 CFR 173: Subpart I) require shipping packages containing radioactive materials to have sufficient radiation shielding to limit the radiation dose rate to 10 millirem per hour at a distance of 2.0 meters (6.6 feet) from the outer lateral surfaces of the transporter. Radioactive material would be released during transportation accidents only if the package carrying the material were subjected to forces that exceeded the package design standard. Only a long-duration severe fire or a powerful collision, both events of extremely low probability, could damage a radioactive material transportation package to the extent that radioactivity could be released to the environment with significant consequences. Type B packages are designed to handle postulated accidents with minimal release of the contents. However, for very severe beyond design accidents (under the higher severity categories i.e., V and VI) a Type B container could be damaged enough to release some of its contents into the environment (NUREG/CR-6672).

Regulations pertaining to the transportation of radioactive materials are primarily published by the U.S. Department of Transportation (DOT) (49 CFR Part 173) and U.S. Nuclear Regulatory Commission (NRC) (10 CFR Part 71). A summary of these regulations are in DOT's *Radioactive Material Regulations Review* (RAMREG-12-2008) (DOT 2008).

In this analysis, both Type A (DUNH) and Type B (spent nuclear fuel from Germany and LEU and LEU/thorium grouted LLW) packages are used. Transportation packaging for radioactive materials are designed, constructed, and maintained to contain and shield its contents during normal transport conditions. The type of packaging used is determined by the total radioactive hazard presented by the material within the packaging. For lower activity materials, Type A packaging is used. For highly radioactive material, such as high-level radioactive waste or spent nuclear fuel, Type B packaging is used. Type A packaging is designed to retain its radioactive contents under normal transportation conditions while Type B packaging is designed to retain its radioactive contents under both normal and accident conditions. Specific requirements for these packages are detailed in 49 CFR Part 173, Subpart I.

Three types of containers would be used to transport radioactive material and waste. **Table 4-26** lists the types of containers evaluated in the analysis along with their volumes, mass, and the number of containers in a shipment. A shipment is defined as the amount of waste transported on a single truck or rail shipment. A rail shipment is defined as 8 rail cars (i.e. 1 rail shipment would consist of 8 rail cars with 2 casks per rail car for a total of 16 casks).

Table 4-26: Material and Associated Shipment Characteristics

Material	Container	Container Volume (cubic feet)[a]	Container Mass (tons)[b]	Shipment Description
SNF from Germany	CASTOR cask (Type B)	22.2	28	8 rail cars with 2 casks per rail car; or 1 cask per truck [c]
LEU or LEU/Thorium (grouted waste)	CASTOR cask (Type B)	22.2	28	8 rail cars with 2 casks per rail car; or 1 cask per truck [c]
LEU or LEU/Thorium (grouted waste)	RH-72B (Type B)	31.4	4	1 cask per truck
Uranyl nitrate (depleted uranium)	55-gallon drum (Type A)	7.35	0.3	72 drums per truck or rail car [d]

LEU = low-enriched uranium; LLW = low-level radioactive waste; SNF = spent nuclear fuel

[a] Container interior volume.

[b] Filled container maximum mass. Container mass includes the mass of the container shell, its internal packaging, and the materials within.

[c] One rail shipment would consist of 8 rail cars with 16 total casks.

[d] Only one shipment consisting of one rail car would be needed to transport this material

Note: To convert from cubic feet to cubic meters, multiply by 0.028317; from pounds to kilograms, by 0.45360.

Source: Laug 1998

In general, the number of shipping containers per shipment was estimated on the basis of the dimensions and weight of the shipping containers, the Transport Index,[37] and the transport vehicle dimensions and weight limits.

4.1.3.5.4 Risk Analysis Results

For transportation accidents, the risk factors are given for both radiological impacts, in terms of potential LCFs in the exposed population, and nonradiological impacts, in terms of number of traffic fatalities. LCFs represent the number of additional latent fatal cancers among the exposed population in the event of an accident. Under accident conditions, the population would be exposed to radiation from released radioactivity if the package were breached and would receive a direct dose (dose received while in close proximity to the outer lateral surfaces of the transport package) if the package were not breached.

Per-shipment risk factors were calculated for the crew and for collective populations of exposed persons for anticipated routes and shipment configurations. Radiological risks are presented in doses per shipment for each unique route, material, and container combination. Radiological risk factors per shipment for incident-free transportation and accident conditions are presented in **Table 4-27**. These factors have been adjusted to reflect the projected population in 2020. For incident-free transportation, both dose and LCF risk factors are provided for the crew and exposed population. The radiological risks would result from potential exposure of people to external radiation emanating from the packaged waste. The exposed population includes residents and car occupants along the route and public at rest stops and fuel stops. The accident radiological risk

[37] The Transport Index is a dimensionless number (rounded up to the next tenth) placed on the label of a package, to designate the degree of control to be exercised by the carrier. Its value is equivalent to the maximum radiation level in millirem per hour at 1 meter (3.3 feet) from the package (10 CFR 71.4 and 49 CFR 173.403).

inherently includes the probability that an accident has occurred that results in the release of radioactive materials.

Although all CASTOR casks do not contain the same amount of radioactive material, for purposes of analysis, it was conservatively assumed that all CASTOR casks contain the inventory that would result in the highest dose.

Table 4-27: Risk Factors for Each Truck or Rail Shipment of Radioactive Material and Waste

Material or Wastes	Origin	Transport Destination	Container	Incident-Free				Accident	
				Crew Dose (person-rem)	LCFs [a]	Population Dose (person-rem)	LCFs [a]	Radiological Risk [a]	Nonradiological Risk (traffic fatalities) [a]
SNF from Germany	JBC	SRS	CASTOR Cask	4.2×10^{-3}	3×10^{-6}	1.9×10^{-2}	1×10^{-5}	2×10^{-14}	3×10^{-5}
LEU or LEU/Thorium	SRS	NNSS (truck)	RH-72B	2.1×10^{-2}	1×10^{-5}	8.0×10^{-3}	5×10^{-6}	1×10^{-14}	2×10^{-4}
LEU or LEU/Thorium	SRS	NNSS (intermodal) [b]	CASTOR Cask	2.4×10^{-2}	1×10^{-5}	2.6×10^{-2}	2×10^{-5}	4×10^{-13}	1×10^{-4}
LEU or LEU/Thorium	SRS	NNSS (intermodal) [b,c]	RH-72B	3.7×10^{-2}	2×10^{-5}	3.2×10^{-2}	2×10^{-5}	7×10^{-16}	2×10^{-4}
LEU or LEU/Thorium	SRS	Commercial disposal facility (truck) [d]	RH-72B	1.9×10^{-2}	1×10^{-5}	7.1×10^{-3}	4×10^{-6}	3×10^{-14}	2×10^{-4}
LEU or LEU/Thorium	SRS	Commercial disposal facility (rail) [d]	CASTOR Cask	8.2×10^{-2}	5×10^{-5}	9.2×10^{-2}	5×10^{-5}	3×10^{-14}	1×10^{-4}
LEU or LEU/Thorium	SRS	Commercial disposal facility (rail) [c,d]	RH-72B	1.2×10^{-1}	7×10^{-5}	1.2×10^{-1}	7×10^{-5}	4×10^{-15}	1×10^{-4}
Uranyl nitrate	AREVA [e]	SRS (rail)	55-gallon drums	1.4×10^{-2}	8×10^{-6}	1.9×10^{-2}	1×10^{-5}	9×10^{-7}	2×10^{-4}
Uranyl nitrate	AREVA [e]	SRS (truck)	55-gallon drums	5.0×10^{-2}	3×10^{-5}	2.7×10^{-2}	2×10^{-5}	5×10^{-6}	2×10^{-4}

DOE = U.S. Department of Energy; JBC = Joint Base Charleston – Weapons Station; LCF = latent cancer fatality; LEU = low-enriched uranium; LLW = low-level waste; NNSS = Nevada National Security Site; SNF = spent nuclear fuel; SRS = Savannah River Site

[a] Risk is expressed in terms of LCFs, except for the nonradiological risk, where it refers to the number of traffic accident fatalities. Radiological risk is calculated for one-way travel while nonradiological risk is calculated for two-way travel. Accident dose-risk can be calculated by dividing the risk values by 0.0006 (DOE 2003). The values are rounded to one non-zero digit.

[b] Intermodal shipments involve transport by a combination of rail and truck. The packages are transferred between rail and truck at an intermodal terminal.

[c] Shipments would only occur under the LEU/Thorium Option to transport the 5,500 cubic feet (160 cubic meters) of LLW that would not fit inside the 455 CASTOR casks.

[d] In order to generate conservative results, it was assumed that, if LEU or LEU/Thorium waste were shipped to a commercial disposal facility (either Energy Solutions in Utah or Waste Control Specialists in Texas), it would be shipped to the site that would result in the larger impacts.

[e] AREVA is located in Richland, Washington.

4.1.3.5.5 Transportation Impacts

Using the number of shipments shown in **Table 4-28** and the per-shipment values from Table 4-27, total risks to the crew and the general population were calculated for each alternative and option. Table 4-28 summarizes transportation risks under each alternative and option considering all shipments of radioactive material and waste.

Table 4-28: Risks from Transporting Radioactive Material and Waste under Each Option and Alternative

Alternative/ Option	Material	Number of Shipments[b]	One-way Distance Traveled (miles)	Incident-Free[a] Crew Dose (person-rem)	Incident-Free[a] Crew LCF Risk[c]	Incident-Free[a] Population Dose (person-rem)	Incident-Free[a] Population LCF Risk[c]	Accident[a] Radiological Risk[c]	Accident[a] Non-radiological Risk[c]
H-Area Alternative									
Vitrification Option	SNF	30	3,900	1.2×10^{-1}	7×10^{-5}	5.4×10^{-1}	3×10^{-4}	5×10^{-13}	9×10^{-4}
LEU Waste Option	SNF	30	3,900	1.2×10^{-1}	7×10^{-5}	5.4×10^{-1}	3×10^{-4}	5×10^{-13}	9×10^{-4}
	DUNH	1	2,500	5.0×10^{-2}	3×10^{-5}	2.7×10^{-2}	2×10^{-5}	5×10^{-6}	2×10^{-4}
	LEU[d]	300	673,000	6.4	4×10^{-3}	2.4	1×10^{-3}	3×10^{-12}	5×10^{-2}
	Total	**330**	**679,000**	**6.5**	**4×10^{-3}**	**3.0**	**2×10^{-3}**	**5×10^{-6}**	**5×10^{-2}**
LEU/Thorium Waste Option	SNF	30	3,900	1.2×10^{-1}	7×10^{-5}	5.4×10^{-1}	3×10^{-4}	5×10^{-13}	9×10^{-4}
	DUNH	1	2,500	5.0×10^{-2}	3×10^{-5}	2.7×10^{-2}	2×10^{-5}	5×10^{-6}	2×10^{-4}
	LEU[d]	510	1,140,000	10.8	6×10^{-3}	4.1	2×10^{-3}	5×10^{-12}	9×10^{-2}
	Total	**540**	**1,140,000**	**10.9**	**7×10^{-3}**	**4.7**	**3×10^{-3}**	**5×10^{-6}**	**9×10^{-2}**
L-Area Alternative	SNF	30	3,900	1.2×10^{-1}	7×10^{-5}	5.4×10^{-1}	3×10^{-4}	5×10^{-12}	9×10^{-4}

DOE = U.S. Department of Energy; DUNH = depleted uranium (uranyl nitrate ([liquid form]); JBC = Joint Base Charleston; LCF = latent cancer fatality; LEU = low enriched uranium, LLW = low-level radioactive waste; NNS = Nevada National Security Site; SNF = spent nuclear fuel; SRS = Savannah River Site

a The totals include impacts from transporting radiological materials and wastes, using destinations that would incur the greatest (most conservative) risks.

b Number of shipments rounded to the nearest ten.

c Risk is expressed in terms of risk of a single LCF, except for the nonradiological risk, where it refers to the risk of a traffic accident fatality. Radiological risk is calculated for one-way travel while nonradiological risk is calculated for two-way travel. Accident dose-risk can be calculated by dividing the risk values by 0.0006 (DOE 2003). The values are rounded to one non-zero digit.

d Disposal at NNSS.

Note: To convert from miles to kilometers, multiply by 1.6093.

The highest risk during incident-free transportation would be under the LEU/Thorium Waste Options, where the risk to the crew would be 7×10^{-3} LCFs and the risk to the public would be 3×10^{-3} LCFs. This risk can also be interpreted to mean that there is approximately 1 chance in 140 that an additional LCF could be experienced among the exposed workers and 1 chance in 330 that an additional LCF could be experienced among the exposed population residing along the transport route.

The highest radiological risk due to an accident would be under the LEU Waste and LEU/Thorium Waste Options, where the risk would be 5×10^{-6} LCFs. This risk can also be interpreted to mean that there is approximately 1 chance in 200,000 that an additional LCF could be experienced as a result of an accident.

The nonradiological accident risk (the potential for fatalities as a direct result of traffic accidents) is greater than the radiological accident risk. The highest risk of a nonradiological accident is 9×10^{-2} under the LEU/Thorium Waste Options. For comparison, in the United States in 2010 there were over 3,900 fatalities due to crashes involving large trucks (DOT 2012a) and over 32,000 traffic fatalities due to all vehicular crashes (DOT 2012b).

Based on this analysis, no fatalities would be expected and the risk to the crew and the general population from the maximum number of shipments associated with the proposed action under all options would be negligible.

4.1.3.6 Environmental Justice

Construction - As indicated in Section 4.1.3.2, workers installing the carbon digestion capability in H-Canyon would receive a small radiation dose. Any additional dose from construction to the MEI and the average offsite individual would be negligible.

Operations - Under the alternatives evaluated in this EA there could be a small additional worker dose during operations, however the additional dose to the MEI and the average offsite individual would be negligible (see Human Health, Section 4.1.3.2). Additionally, the dose to the general population from the maximum number of potential shipments associated with the alternatives would be negligible (see Transportation, Section 4.1.3.6).

Consequently, the dose to any offsite individual would not result in an appreciable increase in the risk of developing an LCF under either alternative evaluated in this EA. Therefore, neither alternative would result in disproportionately high and adverse impacts on minority and low-income populations.

4.1.3.7 Other Resource Areas

4.1.3.7.1 Land Resources – Land Use

The predominant impacts on land use would result from land disturbance from construction activities under the Proposed Action. Under the H-Area Alternative, total land disturbance in H-Area would be approximately 5.4 acres (2.2 hectares): less than 0.4 acres (0.16 hectares) for the additional storage pads, and an additional 5 acres (2 hectares) of land may be disturbed in H-Area under the LEU Waste and LEU/Thorium Waste Options for the construction of the uranium solidification facility (DOE 2014a). Total land disturbance in L-Area would be less than 2.7 acres (1.1 hectares): 1.7 acres (0.7 hectares) for the addition of crushed stone roads and cask storage pads, and another 1 acre (0.4 hectare) for the new sand filter under the L-Area Alternative (DOE 2014a). In both areas, the existing land use would not change. H-Area and L-Area are both industrial areas that have been disturbed in the past, so land use would not be appreciably altered by the proposed activities. Therefore, impacts on land use would be minor and are not discussed further.

4.1.3.7.2 Land Resources – Visual Resources

Impacts are related to construction of new facilities or modifications to existing facilities that may affect visual resources.

As discussed in Section 4.1.3.7.1, construction activities under the Proposed Action not occurring within existing facilities would be limited to road improvements, access road installation, construction of cask storage pads, and construction of the uranium solidification facility at H-Area (only under H-Area Alternative LEU Waste and LEU/Thorium Waste Options). Additionally, commercially-designed and fabricated weather enclosures (steel superstructure with fabric cover) may be installed over the cask storage pads. Fencing and traffic barriers (e.g. concrete dividers or Jersey barriers) may be required for additional security and to minimize vehicle impacts.

In each case, and at each location, the proposed construction activities and associated operations would occur in cleared areas proximate to existing industrial uses at those locations. These activities are consistent with historical activities associated with the two facilities. Additionally, the construction and operation activities would not be visible from off-site locations. Accordingly, the potential construction and operation activities would present a minimal impact to visual resources. These developed areas are consistent with a Visual Resource Management Class IV designation; the proposed new or modified facilities would not change that designation. Therefore, impacts on visual resources would be minor and are not discussed further.

4.1.3.7.3 Geology and Soils

Impacts on geology and soils can occur from disturbance of geologic and soil materials during land clearing, grading, and excavation activities, and the use of geologic and soils materials during facility construction and operations. Disturbance of geologic and soil materials includes excavating rock and soil, soil mixing, soil compaction, and covering geologic and soil materials with building foundations, parking lots, roadways, and fill materials. Geologic and soil materials used as fill during building and road construction include crushed stone, sand, gravel, and soil.

H-Area and L-Area are both industrial areas that have been disturbed in the past. Under both the H-Area and L-Area Alternatives, the construction of crushed-stone roads and crushed-stone storage pads would be the only activities with potential impacts on geology and soils. As noted in Section 4.3.1.8.1, the surface area potentially impacted by the construction of storage pads and access roads is small. Total land disturbance in H-Area would be less than 0.4 acres (0.16 hectares) for the additional storage pads, requiring less than 20,000 cubic feet of crushed stone. Under the LEU and LEU/Thorium Waste Options, an additional 5 acres (2 hectares) would be disturbed with the construction of the uranium solidification facility. Total land disturbance in L-Area would be less than 1.7 acres (0.7 hectares) for the addition of crushed stone roads and cask storage pads, requiring less than 100,000 cubic feet of crushed stone. An additional 1 acre (0.4 hectare) would be disturbed for the construction of the sand filter, requiring approximately 5,400 cubic feet of crushed stone, sand and gravel (DOE 2014a). Therefore, impacts on geology and soils would be minor and are not discussed further.

4.1.3.7.4 Water Resources

Potential impacts on water resources would be associated with:

- Degradation or impairment of water resource quantity or quality (introduction of chemical materials or sediments into the water column);
- Land use changes that alter water courses, system recharge, drainage patterns, and/or exceed the capacity of existing stormwater management systems; and
- Increases in water consumption that may compromise the availability of water.

Construction activities that have the potential to influence water resources at SRS are limited to proposed access road improvements, installation of new crushed-stone access roads, installation of new cask storage pads, and construction of the uranium solidification facility. The proposed construction activities have the potential to affect the discharge of stormwater runoff and sediments. However, compliance with the existing South Carolina NPDES General Permit (SCR100000) to develop and implement a stormwater pollution prevention plan (SWPPP) for such construction would limit the extent and duration of the impacts. The SWPPP would identify site-specific Best Management Practices (BMPs) designed to minimize impacts from runoff, soil erosion, sedimentation, and construction-related accidental spills and effluent releases. There would be no direct release of contaminated effluents during the road and storage pad construction, and no changes to stream channels, aquatic habitats, or surface water flow or consumption are proposed. Therefore, impacts on water resources would be minor and are not discussed further.

4.1.3.7.5 Noise

As noted in Section 3.3.1, Meteorology, Air Quality and Noise, most industrial facilities at SRS are far enough from the site boundary that noise levels at the boundary would not be measureable or would be barely distinguishable from background levels. Construction and operation activities proposed in support of either alternative would be conducted in existing facilities, or within new structures constructed adjacent to existing facilities. None of these activities are expected to substantially contribute to site noise. Therefore, impacts on noise would be minor and are not discussed further.

4.1.3.7.6 Ecological Resources

This section addresses potential impacts on ecological resources, including terrestrial and aquatic resources, wetlands, and threatened and endangered species. Impacts on ecological resources are generally related to land disturbance activities that could occur during construction; little or no impacts would occur during operations under either alternative. Ecological resources would not be further affected because additional land would not be disturbed during facility operations, and any artificial lighting and noise-producing activities would occur in areas that are already in industrial use. Therefore, this section only describes the impacts from construction.

The physical disturbance of land under both the H-Area and L-Area Alternatives would be limited to minor activities involving installation and improvements of existing roads to facilitate receipt and transport of CASTOR casks and installation of storage pads for CASTOR cask storage in either H-Area or L-Area, or installation of a uranium solidification facility in H-Area. These areas have been previously disturbed and used for industrial purposes for many decades. Total land disturbance in H-Area would be approximately 5.4 acres (2.2 hectares) (DOE 2014a). Total land disturbance in L-Area would be less than 2.7 acres (1.1 hectares). No construction is proposed that would require excavation or conversion of undeveloped land. All construction would be conducted consistent with the *Natural Resources Management Plan for the Savannah River Site* (DOE 2005b). Therefore, there is little potential for the proposed action to impact ecological resources, and impacts on ecological resources are not discussed further.

4.1.3.7.7 Cultural Resources

The physical disturbance of land at SRS under either alternative would be limited to areas in H-Area or L-Area that have been previously disturbed and used for industrial purposes for many decades. No new construction is proposed that would require excavation or conversion of undeveloped land.

Because of the limited construction activities, no impacts on cultural or paleontological resources are expected. Should any such resources be unexpectedly encountered during construction of access roads and storage pads, the Savannah River Archaeological Research Program would be contacted to document the find and determine whether additional recovery and mitigation may be required.

4.1.3.7.8 Infrastructure

This section summarizes potential impacts on the SRS infrastructure, specifically the basic resources and services (e.g., utilities) necessary to support continued operations of existing facilities. Impacts on the waste management infrastructure are discussed in Section 4.1.3.5, Waste Management.

Potential impacts on SRS infrastructure could occur as a result of construction activities. Construction associated with the proposed action alternatives and options would be limited to the improvement of existing roads, construction of new crushed-stone access roads, construction of cask storage pads, construction of the uranium solidification facility, and construction of a new sand filter and stack. Electricity requirements to support access road and storage pad construction and improvements, would be met by portable generators. The road and storage pad construction is anticipated to use less than 6,000 gallons of gasoline and less than 6,000 gallons of diesel fuel (DOE 2014a).

Table 4-29 presents the available infrastructure capacity and projected infrastructure requirements for the H-Area and L-Area Alternatives during operations. Table 4-29 also presents FY2014 H-Canyon and L-Area usage for comparison. As shown in Table 4-29, SRS has substantial available infrastructure capacity. Neither the H-Area nor the L-Area Alternatives would substantially impact available capacity during construction and operations. In addition, between 11,000 and 14,000 gallons of diesel fuel would be used each year to power cranes and transportation equipment. Diesel fuel can be provided as needed, therefore, the SRS infrastructure would not be affected.

Table 4-29: Comparison of Available Infrastructure Capacity, Recent Usage, and Alternatives Requirements

				Alternatives[a]		
		FY2014 Consumption		H-Area		L-Area
Resource	Available Capacity	H Canyon	L-Area	Vitrification Option	LEU Waste and LEU/Thorium Waste Options	Melt and Dilute Option
Electricity (megawatt hours per year)	4,100,000	19,241	9,988	27,000	23,000	15,000
Steam (thousand pounds/year)	NA	68,315	13,949	47,000	57,000	18,000
Water (million gallons per year)	2,630	89[b]	259[b]	72	89	37

[a] Estimates of electricity, steam, and water requirements during operation were developed based on DOE projections provided in the following documents: the *Final Supplemental Environmental Impact Statement, Defense Waste Processing Facility* (DOE 1994b); the *Savannah River Site, Salt Processing Alternatives Final Supplemental Environmental Impact Statement* (DOE 2001a), and DOE 2014a.

[b] Process water only

4.2 NO ACTION ALTERNATIVE

Under the No Action Alternative, the spent nuclear fuel containing U.S.-origin HEU from the AVR and THTR would not be transported to the United States for management and disposition. The spent nuclear fuel would remain in storage in Germany. Because DOE would not undertake any actions involving the global commons, Joint Base Charleston – Weapons Station, and SRS under the No Action Alternative, there would be no incremental impacts on these areas.

4.3 CUMULATIVE IMPACTS FROM THE PROPOSED ACTION

CEQ regulations define cumulative impacts as the effects on the environment that result from implementing the Proposed Action or any of its alternatives when added to other past, present, and reasonably foreseeable future actions, regardless of what agency or person undertakes the other actions (40 CFR 1508.7). Thus, the cumulative impacts of an action can be viewed as the total impact on a resource, ecosystem, or human community of that action and all other activities affecting that resource irrespective of the source.

Cumulative impacts were assessed by combining the effects of alternative activities evaluated in this EA with the effects of other past, present, and reasonably foreseeable actions in the ROI. Many of these actions occur at different times and locations and may not be truly additive. For example, actions affecting air quality occur at different times and locations across the ROI; therefore, it is unlikely that the impacts would be completely additive. The effects were combined irrespective of the time and location of the impact, to envelop any uncertainties in the projected activities and their effects. This approach produces a conservative estimation of cumulative impacts for the activities considered.

Because acceptance of spent nuclear fuel from Germany would cause little to no impacts on land resources, geology and soils, water resources, noise, ecological resources, cultural resources, and infrastructure, acceptance of the spent nuclear fuel from Germany would not result in additional cumulative impacts for these resource areas. Thus, this cumulative impacts section analyzes air quality, human health, socioeconomics, waste management, transportation, and environmental justice associated with transport and disposition of the spent nuclear fuel from Germany.

4.3.1 Global Commons and Joint Base Charleston

Activities that may add to cumulative impacts on the global commons and at Joint Base Charleston – Weapons Station include:

- The ongoing movement of ships carrying radioactive materials across the global commons for general commerce;
- The transportation of FRR SNF to the United States under the FRR SNF acceptance policy; and
- The transportation of HEU and plutonium to the United States under programs to secure fissile material.

Each year there are several million worldwide shipments of radioactive materials using trucks, trains, ocean vessels, aircraft, and other conveyances, including numerous shipments across the global commons. Incident-free transport of the radioactive materials to ports of entry in the United States would not result in radiation exposures to members of the general public. Only the crews of ships carrying the containers of radioactive materials and the dock handlers unloading the containers would be exposed to radiation. Therefore, there would be no cumulative impacts on the public from radiation exposure.

Cumulative radiation doses and risks to crews and dock handlers for transport of radioactive materials from foreign countries to U.S. seaports are summarized in **Table 4-30**. This table includes the doses and risks from shipments of: (1) FRR SNF by ocean vessel under the FRR SNF Acceptance Program (DOE 1996a, 2009a); (2) 5 metric tons (5.5 tons) of HEU by ocean vessel as evaluated in the 2006 *Supplement Analysis for the Air and Ocean Transport of Enriched Uranium between Foreign Countries and the United States* (DOE 2006c); and (3) 100 kilograms (220 pounds) of Gap material plutonium by ocean vessel as evaluated in the *Environmental Assessment of the Receipt and Storage of Gap Material – Plutonium and Finding of No Significant Impact* (DOE 2010b).

The December 2015 *Environmental Assessment for Gap Material Plutonium – Transport, Receipt, and Processing* (DOE 2015e) and Finding of No Significant Impact (DOE 2015f) addressed the receipt of 900 kilograms (1,984 pounds) of plutonium through the Joint Base Charleston – Weapons Station. The additional impacts of these activities have not been quantitatively addressed in this draft EA, but are expected to be a relatively small addition to the cumulative doses and risks to the ships' crews and dock handlers. The additional impacts will be included in the final version of this EA.

Cumulative radiation doses and risks to ship crews and dock handlers from transport of radioactive materials from foreign countries to United Sates seaports would result in a dose of 89 person-rem and no LCFs (calculated value of 0.05). Shipments of spent nuclear fuel from Germany to the United States would represent fractions of the total cumulative dose and risk from transport of radioactive material from foreign countries.

Table 4-30: Cumulative Radiation Doses and Risks for Incident-Free Marine Transport of Radioactive Materials to United States Seaports

Risk Receptor (scenario)		Radiation Dose (person-rem)	Risk (LCFs)
Ship crew, FRR SNF [a]		75.4	0 (5×10^{-2})
Dock handlers, FRR SNF [a]		8.2	0 (5×10^{-3})
Ship crew, 5,000 kilograms of unirradiated HEU [b]		0.030	0 (2×10^{-5})
Dock handlers, 5,000 kilograms of unirradiated HEU [b]		0.13	0 (8×10^{-5})
Ship crew, 100 kilograms of gap material plutonium [c]		1.4	0 (8×10^{-4})
Dock handlers, 100 kilograms of gap material plutonium [c]		0.67	0 (4×10^{-4})
Draft EA action alternatives	Ship crew, spent nuclear fuel from Germany	2.9	0 (2×10^{-3})
	Dock handlers, spent nuclear fuel from Germany	0.24	0 (1×10^{-4})
Totals		**89**	**0** (**0.05**)

EA = environmental assessment; FRR = foreign research reactor, HEU = highly enriched uranium, LCF = latent cancer fatality, SNF = spent nuclear fuel.

[a] Assuming a radiation dose rate of 10 millirem per hour at 2 meters (6.6 feet) from the packaging's surface. Includes shipment of gap material SNF (DOE 2009a). The dose-to-LCF factor assumed in the *FRR SNF EIS* (DOE 1996a) was updated to 0.0006 LCFs per person-rem (DOE 2003).

[b] Additionally assessed was the option of shipping the same 5,000 kilograms of unirradiated HEU by military cargo or commercial aircraft. Air shipment of all unirradiated HEU was projected to result in a collective dose to air crew members of up to 1.1 person-rem and a collective dose to ground cargo workers of up to 0.51 person-rem. The calculated risk values were 7×10^{-4} LCF and 3×10^{-4} LCF, respectively (DOE 2006d).

[c] The *Environmental Assessment for U.S. Receipt and Storage of Gap Material - Plutonium and Finding of No Significant Impact* addressed ship transport and aircraft transport (DOE 2010b); only the ship transport alternative is included here. Conservatively assuming a dose rate of 10 millirem per hour at 2 meters (6.6 feet) from the packaging's surface. The listed impacts reflect an assumed 10 shipments of gap material plutonium by chartered vessel.

Note: Totals may not add due to rounding. To convert kilograms to pounds, multiply by 2.205.

Risks were determined using a dose-to-risk factor of 0.0006 LCFs per person-rem and are presented using 1 significant figure (DOE 2003).

Source: DOE 1996a, 2006c, 2009a, 2010b.

4.3.2 Savannah River Site

In addition to the alternatives evaluated in this Draft EA, actions that may contribute to cumulative impacts at SRS include onsite and offsite projects conducted by Federal, state, and local governments; the private sector; or individuals that are within the ROIs of the actions considered in this Draft EA. Information on present and future actions was obtained from a review of site-specific actions and NEPA documents to determine if current or proposed projects could affect the cumulative impacts analysis at the potentially affected sites. For those actions that are not yet well defined or are not expected to represent meaningful contributions to cumulative impacts, the actions are described but not included in the determination of cumulative effects. The potentially cumulative actions discussed here are the major projects that may contribute to cumulative impacts on or in the vicinity of the potentially affected sites.

4.3.2.1 U.S. Department of Energy Actions

Savannah River Site Salt Processing Alternatives Final Supplemental Environmental Impact Statement (*Salt Processing EIS*) (DOE/EIS-0082-S2) (DOE 2001a). A process to separate the high-activity and low-activity waste fractions in HLW solutions is planned to replace the in-tank precipitation process evaluated in the *Defense Waste Processing Facility Supplemental Environmental Impact Statement* (DOE 1994b). The *Salt Processing EIS* evaluates four alternatives: (1) small tank precipitation; (2) ion exchange; (3) solvent extraction; and (4) direct disposal in grout. The cumulative impacts analysis in this Draft EA includes the maximum impacts of the solvent extraction process, as selected in the DOE Record of Decision (ROD) for the *Salt Processing EIS* (66 FR 52752). On January 24, 2006, DOE issued a revised ROD (71 FR 3834) adopting an approach that implements interim salt processing until the solvent extraction process becomes operational.

Savannah River Site High-Level Waste Tank Closure Final Environmental Impact Statement (*HLW EIS*) (DOE/EIS-0303) (DOE 2002). DOE proposes to close the HLW tanks at F- and H-Areas at SRS in accordance with applicable laws and regulations, DOE orders and regulations, and the Industrial Wastewater Closure Plan for the F- and H-Area High-Level Waste Tank Systems (approved by SCDHEC), which specifies the management of residuals as waste incidental to reprocessing. The proposed action would begin after bulk waste removal has been completed. The *HLW EIS* evaluates three alternatives regarding the HLW tanks at SRS: (1) the Stabilize Tanks Alternative (referred to as the "Clean and Stabilize Tanks Alternative" in the *Draft HLW EIS*, (2) the Clean and Remove Tanks Alternative, and (3) the No Action Alternative. Under the Stabilize Tanks Alternative, the *HLW EIS* considers three options for tank stabilization: Fill with Grout (Preferred Alternative), Fill with Sand, and Fill with Saltstone. Under each alternative (except No Action), DOE would close 49 HLW tanks and associated waste-handling equipment, including evaporators, pumps, diversion boxes, and transfer lines. In the ROD issued on August 19, 2002 (67 FR 53784), DOE selected the Preferred Alternative identified in the *HLW EIS*, Stabilize Tanks—Fill with Grout.

In a 2012 supplement analysis (DOE 2012c), DOE addressed the potential environmental impacts from using additional tank cleaning technologies rather than those specifically analyzed in the *HLW EIS*, and from performing an evaluation using criteria specified in Section 3116(a) of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (Public Law 108-375) rather than the waste incidental to reprocessing criteria specified in DOE Manual 435.1-1, *Radioactive Waste Management*. In a 2014 supplement analysis (DOE 2014d), DOE proposed to make changes to the tank closure process for the F-Area and H-Area Tank Farms. The changes involved projects and technical proposals evaluated in the HLW EIS and the 2012 supplement analysis that have been modified or suspended, and new processes have been developed based on lessons learned from previous tank closures. Most importantly, new performance assessments were prepared for the tank farms. DOE determined that these proposed actions did not constitute substantial changes from those evaluated in the *HLW EIS*, and that no significant new information was identified that would affect the basis for its original decision as documented in the ROD (DOE 2012c, 2014d). DOE closed Tanks 17 and 20 in 1997, Tanks 18 and 19 in 2012, and Tanks 5 and 6 in 2013 (DOE 2014d).

Draft Environmental Impact Statement for the Disposal of Greater-Than-Class C (GTCC) Low-Level Radioactive Waste and GTCC-Like Waste (Draft GTCC EIS) (DOE/EIS-0375-D) (DOE 2011d). In February 2011, DOE issued the Draft GTCC EIS to evaluate the potential environmental impacts associated with the proposed development, operation, and long-term management of a facility or facilities for disposal of greater-than-Class C (GTCC) LLW and DOE GTCC-like waste. GTCC LLW has radionuclide concentrations exceeding the limits for Class C LLW established by NRC in 10 CFR Part 61. The Draft GTCC EIS also considers DOE waste having similar characteristics. Currently, there is no location for disposal of GTCC LLW and the Federal government is responsible for such disposal under the Low-Level Radioactive Waste

Policy Amendments Act of 1985 (Public Law 99-240). Section 631 of the Energy Policy Act of 2005 requires DOE to submit a report to Congress on disposal alternatives under consideration and await Congressional action before making a final decision on which disposal alternative to implement. SRS is one of the six candidate DOE sites being considered for GTCC LLW disposal in the Draft GTCC EIS, which also include Hanford, Idaho National Laboratory, Los Alamos National Laboratory, Nevada National Security Site, and WIPP. DOE is also considering two disposal locations in the WIPP vicinity and generic commercial sites in four regions of the country. DOE is evaluating several disposal technologies in the Draft GTCC EIS, including a geologic repository, intermediate depth boreholes, enhanced near-surface trenches, and above-grade vaults. Enhanced near-surface trenches and above-grade vaults are considered at SRS. Prior to implementation of any alternative examined in the Draft GTCC EIS, follow-on site specific NEPA review would be conducted as appropriate, to identify the location or locations within a given site for a geologic repository, intermediate depth borehole, trench, or vault facility for the disposal of GTCC LLW and GTCC-like wastes.

Final Long-Term Management and Storage of Elemental Mercury Environmental Impact Statement (*Mercury Storage EIS*) (DOE/EIS-0423) (DOE 2011e). The proposed action analyzed in this EIS is the long-term storage of up to 10,000 metric tons (11,000 tons) of elemental mercury within either existing or new facilities at one of seven sites throughout the United States, including SRS. At SRS, a new facility was proposed that would occupy 7.6 acres (3.1 hectares) of the approximately 330-acre (134-hectare) E-Area. The preferred alternative in the *Mercury Storage EIS* was the construction of a new facility at the Waste Control Specialists, LLC, site located near Andrews, Texas; implementing this alternative would result in no cumulative impacts at SRS. However, since publication of the Mercury Storage EIS, DOE has reconsidered the range of alternatives and has issued a *Final Long-Term Management and Storage of Elemental Mercury Supplemental Environmental Impact Statement* (*Final Mercury Storage Supplemental EIS*) (DOE/EIS-0423-S1) to consider three additional locations at or near WIPP (DOE 2013d); the preferred alternative is unchanged.

Environmental Assessment for the Proposed Use of the Savannah River Site Lands for Military Training (DOE/EA-1606) (DOE 2011c). DOE prepared this environmental assessment to evaluate potential environmental impacts regarding the use of SRS by the U.S. Departments of Defense and Homeland Security (DOD and DHS, respectively) for military training purposes. Alternatives considered are No Action (i.e., SRS would not be used for military training) and the proposed action (i.e., use of a specific area of SRS for non-live-fire tactical maneuver training). The purpose of the proposed action is to enable DOD and DHS to conduct low intensity, non-live-fire tactical maneuver training activities on SRS to support current and future mission requirements. Based on the analyses in the environmental assessment, DOE determined that the proposed action is not a major Federal action significantly affecting the quality of the human environment within the meaning of NEPA and issued a Finding of No Significant Impact (DOE 2011c).

Supplement Analysis, Savannah River Site Spent Nuclear Fuel Management Environmental Impact Statement (*SRS SNF Management EIS*) (DOE/EIS-0279-SA-01 and DOE/EIS-0218-SA-06) (DOE 2013e). In this supplement analysis, DOE evaluated the impacts of managing a limited quantity of spent nuclear fuel using conventional processing rather than melt and dilute technology. In addition, DOE evaluated the receipt and processing of HEU target residue materials from the Chalk River Laboratories in Canada. DOE concluded that the impacts of these actions were addressed in the *SRS SNF Management EIS*. H-Canyon operations are included in the baseline impacts of ongoing SRS operations. Therefore, this activity would not substantially contribute to increased cumulative impacts at SRS.

Supplement Analysis For the Foreign Research Reactor Spent Nuclear Fuel Acceptance Program: Highly Enriched Uranium Target Residue Material Transportation (DOE/EIS-0218-SA-07) (DOE 2015d) DOE prepared this supplement analysis to reflect information obtained since issuance of the 2013 supplement analysis for the *SRS SNF Management EIS* (DOE 2013e) and bearing on the potential impacts that could result from transporting HEU target residue material from Canada to the United States. This supplement analysis supported DOE's determination in the 2013 supplement analysis (DOE 2013e) that the impacts associated with transport of the material would be very low. Nothing was identified indicating a need to reassess DOE's conclusions in the 2013 supplement analysis (DOE 2013e) with respect to either transport of the material or its disposition at SRS.

Final Surplus Plutonium Disposition Supplemental Environmental Impact Statement (*SPD Supplemental EIS*) (DOE/EIS-0283-S2) (DOE 2015a). The *SPD Supplemental EIS* addressed disposition of an additional 13.1 metric tons (14.4 tons) of surplus plutonium composed of 7.1 metric tons (7.8 tons) of plutonium from pits and 6 metric tons (6.6 tons) of non-pit plutonium. In addition to fabrication of surplus plutonium into MOX fuel at MFFF, the action alternatives addressed disposition pathways where surplus plutonium would be immobilized using a new vitrification capability at K-Area followed by vitrification with HLW at the DWPF in S-Area; dissolved at the H-Canyon/HB-Line followed by vitrification at DWPF; or prepared at SRS facilities such as H-Canyon/HB-Line for disposal as CH-TRU waste at WIPP. Canisters of HLW from DWPF would be stored in S-Area pending their disposition. Finally, the *SPD Supplemental EIS* evaluated the impacts of options for disassembly and conversion of the pit plutonium. These options included use of newly constructed and existing facilities at SRS and at the Los Alamos National Laboratory (DOE 2015a).

DOE did not identify a Preferred Alternative in the April 2015 *SPD Supplemental EIS* (DOE 2015a). On December 24, 2015, DOE announced a Preferred Alternative for the 6 metric tons (6.6 tons) of surplus non-pit plutonium (80 FR 80348). DOE's Preferred Alternative is to prepare this plutonium in H-Canyon/HB-Line or the K-Area Complex at SRS for eventual disposal at WIPP near Carlsbad, New Mexico. DOE has no Preferred Alternative for the disposition of the remaining 7.1 metric tons (7.8 tons) of surplus plutonium from pits, nor does it have a Preferred Alternative among the pathways analyzed for providing the capability to disassemble surplus pits and convert the plutonium from pits to a form suitable for disposition. DOE may issue a Record of Decision no sooner than 30 days after its announcement of a Preferred Alternative (DOE 2015a).

Environmental Assessment for Gap Material Plutonium – Transport, Receipt, and Processing (DOE/EA-2024) (DOE 2015e). This environmental assessment evaluated the potential environmental impacts of transporting up to 900 kilograms (1,984 pounds) of plutonium from foreign countries to SRS for storage and processing pending final disposition. DOE will transport packaged plutonium by ship from foreign countries to Joint Base Charleston –Weapons Station in South Carolina, transfer the packages to a specially designed truck transporter, transport the materials to SRS, and place the plutonium into an approved storage facility in K-Area. Gap material plutonium will be stabilized in a capability to be installed in H-Canyon/HB-Line or the K-Area Complex at SRS. On December 28, 2015, DOE/NNSA issued a Finding of No Significant

Impact (DOE 2015f) for the proposed action. Gap material plutonium would be dispositioned along with U.S. surplus plutonium as described in the *SPD Supplemental EIS* (DOE 2015a). The cumulative impacts of activities for Gap material plutonium management at SRS are expected to be small and will be included in the final version of this EA.

Impact on SRS Site Closure. DOE's Office of Environmental Management has on ongoing missions at SRS to remediate and clean up the legacy of nuclear materials production from the 1950s through the 1980s. Although 85 percent of the industrial footprint has been cleaned up and remediated for potential reuse or development, cleanup operations of major nuclear facilities supporting disposition of liquid waste and surplus weapons plutonium will continue for several more decades. The Environmental Management cleanup program, involving stabilization and disposition of nuclear materials, disposition of liquid waste, and tank closure, is expected to continue through FY2042.

The National Nuclear Security Administration also has ongoing missions at SRS in support of stockpile stewardship and management and materials disposition. These tritium operations and other stockpile stewardship activities are enduring missions that will last well beyond the Environmental Management cleanup program (SRNS 2014c).

Construction, modification, and operation of the facilities that would be used to disposition the spent nuclear fuel from Germany is not expected to impact resources associated with current or future site activities, remediation efforts or site closure. Because Germany would pay for disposition of its spent nuclear fuel, U.S. government funding for other SRS projects would not be affected. A solidification facility and storage pads in H-Area, or a sand filter, fan room, stack, new truck bay, and storage pads in L-Area would be the only new construction required if this project were implemented. Most of the activities would be performed in existing facilities that would require varying degrees of modification, none of which would impact future decommissioning, decontamination and demolition efforts to an appreciable degree. Therefore, the impact of the proposed project on site closure, if any, would be the additional time facilities would operate before they could be decommissioned. As indicated in the following paragraphs, the maximum impact on SRS site closure is estimated to be 1 year (DOE 2014a).

> **H-Area Alternative, Vitrification Option.** Carbon digestion and kernel dissolution operations could occur at the same time as other H-Canyon operations for managing materials currently at SRS. However, the digestion and dissolution process would result in additional wastes to be processed through DWPF and the saltstone facilities, which would add approximately 100 days to DWPF operations and 24 days to saltstone facilities operations if this option were implemented.

> **H-Area Alternative, LEU or LEU/Thorium Waste Option.** Carbon digestion and kernel dissolution operations could occur at the same time as other H-Canyon operations for managing materials currently at SRS. However, under either of these options, solvent extraction operations for the spent nuclear fuel from Germany would occur after all other scheduled materials had been processed through H-Canyon. This material would be the last campaign in H-Canyon and would result in H-Canyon and associated facilities operating approximately 1.5 years longer than currently projected. These materials could be processed while de-inventorying and deactivating activities occurred in other parts of H-Canyon. Because actions supporting decommissioning could proceed in other parts of

H-Canyon, the effect of processing the spent nuclear fuel from Germany on H-Canyon decommissioning would be an extension of approximately 1 year for either of these options. The digestion and dissolution process would result in additional wastes to be processed through DWPF and the saltstone facilities, which would add approximately 30 days to both DWPF operations and saltstone facilities operations if this option were implemented.

L-Area Alternative. Carbon digestion and kernel dissolution operations could occur at the same time as, but independent of the processing of other SRS spent fuel inventory. The melt and dilute process could also use a large portion of the current SRS spent fuel inventory. However, the melt and dilute process would result in additional wastes to be processed through the saltstone facilities, which would add approximately 16 days of operational time to these facilities if this alternative were selected.

4.3.2.2 Other Actions

Nuclear facilities in the vicinity of SRS that may contribute to cumulative impacts at SRS include Georgia Power's two-unit Vogtle Electric Generating Plant across the river from SRS; Energy*Solutions'* commercial LLW disposal facility just east of SRS; and Starmet CMI, Inc. (formerly Carolina Metals), located southeast of SRS, which processes uranium-contaminated metals. The Vogtle Plant, the Energy*Solutions* facility, and the Starmet CMI facility are located approximately 11, 8, and 15 miles (18, 13, and 24 kilometers), respectively, from the center of SRS. NRC has issued the *Final Supplemental Environmental Impact Statement for Combined Licenses (COLs) for Vogtle Electric Generating Plant Units 3 and 4* (NRC 2011) addressing two additional units at the Vogtle Plant, and has approved the combined construction and operating license for both units (NEI 2012). Due to the proximity of the plant to SRS, the cumulative impacts of expansion of the Vogtle Plant are addressed for each resource area, as appropriate. Annual monitoring reports filed with the State of South Carolina indicate that operation of the Energy*Solutions* facility and the Starmet CMI facility does not noticeably affect radiation levels in air or water in the vicinity of SRS. Therefore, they are not included in this assessment. Other nuclear facilities (e.g., Virgil C. Summer Nuclear Station, Unit 1, operated by South Carolina Electric and Gas) are too far (more than 50 miles [80 kilometers]) from SRS to have an appreciable cumulative effect (DOE 2002).

Numerous existing and planned industrial facilities (e.g., textile mills, paper product mills, and manufacturing facilities) operate or are anticipated to operate within the counties surrounding SRS, with permitted air emissions and discharges to surface waters. Because of the distances between SRS and these private industrial facilities, there is little opportunity for interaction of plant emissions, and no major cumulative impacts on air or water quality are expected (DOE 2002).

An additional offsite facility having the potential to affect the nonradiological environment is South Carolina Electric and Gas Company's Urquhart Station. Urquhart Station is a three-unit, 250-megawatt, coal- and natural gas-fired steam electric plant in Beech Island, South Carolina, located about 18 miles (29 kilometers) north of SRS. Because of the distance between SRS and Urquhart Station, and the regional wind direction frequencies, there is little opportunity for any interaction of plant emissions, and no major cumulative impacts on air quality are expected (DOE 2002).

4.3.2.3 Cumulative Impacts on Air Quality

Effects on air quality from construction and operations activities at SRS could result in temporary increases in air pollutant concentrations at the site boundary. Construction impacts would be similar to the impacts that would occur during construction of a similar-sized housing development or a commercial project. Emissions of fugitive dust from these activities would be controlled using water sprays and other engineering and management practices, as appropriate. Because construction activities would be minor and small areas of land would be disturbed, air quality impacts would be minor and are not likely to substantially contribute to cumulative impacts.

Much of the operations activities would utilize existing processes and equipment and therefore would not result in additional incremental air quality impacts. For new processes, pollutant control measures such as scrubbers, filters, and other control technologies would ensure that pollutant emissions are minimal and within current regulatory thresholds. The maximum ground-level concentrations off site and along roads to which the public has regular access would be below ambient air quality standards. Because the operation of facilities for processing spent nuclear fuel from Germany would produce relatively small quantities of criteria air pollutants and hazardous air pollutants, these emissions are not likely to substantially contribute to cumulative impacts.

4.3.2.4 Cumulative Impacts on Human Health

Cumulative radiological health effects on the public in the vicinity of SRS are presented in terms of radiological doses, associated excess LCFs in the offsite population, and associated LCF risk to a hypothetical MEI. Radiological health effects on involved SRS workers are presented in terms of radiological doses and associated excess LCFs in the workforce. **Table 4-31** summarizes the annual cumulative radiological health effects from routine SRS operations, proposed DOE actions, and non-Federal nuclear facility operations. The Vogtle Electric Generating Plant, located in Waynesboro, GA, is used as the representative non-Federal nuclear facility.

Table 4-31: Annual Cumulative Population Health Effects of Exposure to Radioactive Contaminants at the Savannah River Site

Activity		Population within 50 Miles (80 kilometers)		MEI	
		Dose (person-rem per year)	Annual LCFs [a]	Dose (millirem per year)	Annual LCF Risk [a]
Past, Present, and Reasonably Foreseeable Future Actions					
Existing site activities (Baseline)[b]		3.4	0 (0.002)	0.12	7×10^{-8}
High-Level Radioactive Waste Salt Processing Facility (DOE 2001a)		18	0 (0.01)	0.31	2×10^{-7}
Tank closure (DOE 2002)		1.4×10^{-3}	0 (8×10^{-7})	2.5×10^{-5}	2×10^{-11}
Disposal of greater-than-Class C low-level radioactive waste (DOE 2011a) [c]		-	-	-	-
Surplus Plutonium Disposition (DOE 2015a) [d]		0.97	0 (6×10^{-4})	0.010	6×10^{-9}
Subtotal - Baseline Plus Other DOE Actions		**22**	**0 (0.01)**	**0.44**	**3×10^{-7}**
German Fuel EA action alternatives [e]	H-Area	7.3 to 7.8	0 (4×10^{-3} to 5×10^{-3})	0.084 to 0.12	5×10^{-8} to 6×10^{-8}
	L-Area	2.3	0 (1×10^{-3})	2.9×10^{-2}	2×10^{-8}
Total for Savannah River Site		**25 to 30**	**0 (0.01 to 0.02)**	**0.47 to 0.56 [f]**	**3×10^{-7}**
Vogtle Plant (NRC 2008, 2011)		1.8	0 (0.001)	2.4	1×10^{-6}
Total for Region		**26 to 32**	**0 (0.02)**	**- [f]**	**- [f]**

EA = environmental assessment; LCF = latent cancer fatality; MEI = maximally exposed individual.

a The annual LCFs for the analyzed population represent the number of LCFs calculated by multiplying the listed doses by the risk conversion factor; no population LCFs are expected from any individual activity or from all combined activities. The annual MEI LCF risk represents the calculated risk of an LCF to an individual.

b Impact indicators are from Chapter 3, Section 3.3.2.1, of this EA, and representing an average for the years 2009 through 2013.

c It is not expected that the general public would receive any measurable radiation doses during waste disposal operations given the solid nature of greater-than-Class C LLW and the distance of potential waste handling activities from potentially affected individuals.

d Values are for the largest doses and risks over all alternatives addressed in the *Final SPD Supplemental EIS* (DOE 2015a).

e Impact indicators are from Section 4.1.3.2.

f The same individual would not be the MEI for all activities at SRS and the Vogtle Plant; therefore, MEI impacts for SRS and the Vogtle Plant have not been summed.

Note: Due to rounding, the column totals may be slightly different than those obtained by summing the individual values.
LCFs and LCF risks are calculated using a factor of 0.0006 LCFs per rem or person-rem (DOE 2003).

As shown in Table 4-31, the annual cumulative offsite population dose is estimated to be 26 to 32 person-rem for the regional population. This annual population dose is not expected to result in any LCFs. Activities proposed under this EA could result in annual doses of 7.3 to 7.8 person-rem under the H-Area Alternative and 2.3 person-rem under the L-Area Alternative with no associated LCFs for either alternative. For perspective, the annual doses to the same local population from naturally occurring radioactive sources (311 millirem per person – see Chapter 3, Section 3.3.2.1) would be about 270,000 person-rem, from which approximately 160 LCFs would be inferred. The assumed population for this estimate, about 860,000 persons in the year 2020, is the average of the populations within 50 miles (80 kilometers) of SRS.

Table 4-31 indicates that the maximum annual dose to the MEI at SRS may be up to 0.56 millirem per year; this dose is much less than applicable DOE regulatory limits (10 millirem per year from the air pathway, 4 millirem per year from the liquid pathway, and 100 millirem per year for all pathways).[38] This is a very conservative estimate of potential dose to an MEI because the SRS activities contributing to this dose are not likely to occur at the same time and location.

Table 4-32 summarizes annual cumulative worker doses and annual LCFs from routine DOE operations and proposed DOE actions at SRS. The maximum cumulative annual SRS worker dose could be up to 870 person-rem, which is not expected to cause an LCF among the involved worker population. From 2009 through 2013, involved workers at SRS received an average annual radiation dose from normal operations of 134 person-rem (see Chapter 3, Section 3.3.2.1). Activities proposed under the action alternatives could result in annual workforce doses of 28 to 41 person-rem under the H-Area Alternative and 8 person-rem under the L-Area Alternative with no LCFs for either alternative. Doses to individual workers would be kept below the regulatory limit of 5,000 millirem per year (10 CFR 835.202). Further, ALARA principles would be implemented to maintain individual worker doses below the DOE Administrative Control Level of 2,000 millirem (DOE 2009b). The SRS ALARA goal is to limit annual individual exposures to 500 millirem (SRS 2014).

Table 4-32: Annual Cumulative Health Effects on Savannah River Site Workers from Exposure to Radioactive Contaminants

Activity		Involved Workers: Dose (person-rem per year)	Involved Workers: Annual LCFs [a]
Past, Present, and Reasonably Foreseeable Future Actions			
Existing site activities for 2010 (Baseline) [b]		134	0.08
High-Level Radioactive Waste Salt Processing Facility (DOE 2001a)		6.5	0.004
Tank Closure (DOE 2002)		53	0.03
Disposal of greater-than-Class C low-level radioactive waste (DOE 2011d)[c]		5.2	0.003
Surplus Plutonium Disposition (DOE 2015a)[d]		630	0.4
Baseline Plus Other DOE Actions		**830**	**0.5**
Draft EA action alternatives[e]	H-Area	28 to 41	0.02
	L-Area	8	0.005
Total [g]		**840 to 870**	**0.5**

EA = environmental assessment; LCF = latent cancer fatality.

[a] LCFs were calculated using a conversion of 0.0006 LCFs per rem or person-rem (DOE 2003). The annual LCFs for the analyzed worker population represent the calculated number of LCFs obtained by multiplying the listed doses by the risk conversion factor.

[b] Impact indicators represent an average for the years 2009 through 2013, including an average of 2,159 workers that had a measurable dose over these years – see Chapter 3, Section 3.3.2.1.

[c] The indicated doses and LCF risks are associated with the vault method of waste disposal at SRS. Doses and risks associated with the trench method of waste disposal at SRS would be smaller.

[d] Values are for the highest doses and risks over all alternatives evaluated in the *Final SPD Supplemental EIS* (DOE 2015a) considering worker doses for construction and operations.

[e] Impact indicators are from Section 4.1.3.2.

[f] Due to rounding, the column totals may be slightly different than those obtained by summing the individual values.

[38] As derived from DOE Order 458.1, *Radiation Protection of the Public and the Environment*.

4.3.2.5 Cumulative Impacts on Socioeconomics

Construction under both the H-Area and L-Area Alternatives would generate mostly new jobs (approximately 1 to 3 percent of the 7,224 persons employed at SRS). Operations under both the H-Area and L-Area Alternatives would preserve existing jobs (including direct and indirect jobs) and potentially create as many as 20 new jobs (less than 1 percent of the 7,224 total employed at SRS) under the LEU Waste and LEU/Thorium Waste Options. By comparison, approximately 220,989 people were employed in the SRS ROI in 2013 (see Section 3.3.3). As a result, there would be no substantial impacts on socioeconomic conditions from any of the alternatives evaluated in this German Fuel EA and no meaningful contribution to cumulative impacts in the ROI.

4.3.2.6 Cumulative Impacts on Waste Management

Table 4-33 lists cumulative volumes of LLW, hazardous waste, and solid nonhazardous waste that would be generated at SRS from all construction and operational activities including the waste that would be generated under the action alternatives evaluated in this EA. Cumulative waste volumes from existing site activities are projected over 30 years, a period of time that exceeds the projected periods of construction or operation of all involved SRS facilities under the action alternatives addressed in this Draft EA. The cumulative waste volumes include possible disposal of GTCC waste at SRS pursuant to the *Draft GTCC EIS* (DOE 2011d). Also, SRS is being considered for use as a military training site; however, negligible waste generation is expected from this action (DOE 2011c).

Table 4-33: Total Cumulative Waste Generation at the Savannah River Site

Activity (duration or reference)		Estimated Waste Generation (cubic meters) LLW	Hazardous Waste	Solid Nonhazardous Waste
Past, Present, and Reasonably Foreseeable Future Actions				
Existing site activities (30 years)[a]		390,000	720	2,310,000
ER/D&D; 35-Year Forecast (DOE 2002)		61,600	3,100 [b]	N/R
HLW Salt Processing Facility[c] (DOE 2001a)		920	43	7,670 [d]
Tank closure (DOE 2002)[e]		1,284	43	428
Biomass cogeneration and heating (30 years) (DOE 2008b)		0	0	438,000[f]
GTCC LLW facilities (DOE 2011d)[g]		250	440	780,000
GTCC LLW disposal at SRS (DOE 2011d)		12,000	0	0
Surplus Plutonium Disposition (DOE 2015a)[h]		9,700 – 34,000	5 to 7,000	13,000 to 45,000
Subtotal - Baseline Plus Other Actions		**476,000 to 500,000**	**4,400 to 11,000**	**3,550,000 to 3,580,000**
Draft EA action alternatives[h]	H-Area	2,400 to 3,000	0.2 to 570	110 to 420
	L-Area	2,400 [i]	0	0
Total		**478,000 to 503,000**	**4,400 to 12,000**	**3,550,000 to 3,580,000**

D&D = decontamination and decommissioning; DWPF = Defense Waste Processing Facility; EA = environmental assessment; ER = environmental restoration; GTCC = greater-than-Class C; HLW = high-level radioactive waste; LLW = low-level radioactive waste; N/R = not reported; SRS = Savannah River Site.

[a] Except for HLW, volumes were obtained from Chapter 3, Section 3.3.4.1, assuming the 5-year average annual generation rate would continue for 30 years. HLW is currently stored in waste storage tanks as discussed in Section 3.3.4.2.

[b] About 6,200 cubic meters of combined mixed LLW and hazardous waste was estimated (DOE 2002); half was assumed to be hazardous waste.

[c] Under the preferred solvent extraction cesium separations process, salt waste processing could also generate about 45,400 cubic meters of liquid radioactive waste that would be evaporated (DOE 2001a).

[d] Assuming 910 metric tons of sanitary solid and industrial waste to be disposed of at the Three Rivers Regional Landfill (DOE 2001a), and a non-compacted waste density of 0.1186 metric tons per cubic meter (200 pounds per cubic yard).

[e] Under the preferred Fill-with-Grout option, tank closure activities could also generate about 48,600 cubic meters of liquid radioactive waste that would be evaporated (DOE 2002).

[f] Assuming 30 years of wood ash generation at a rate of about 7,300 metric tons per year (DOE 2008b), and a wood fly ash density of 490 kilograms per cubic meter (31 pounds per cubic foot) (Naik 2002).

[g] Highest potential construction and operations generation volume from either the trench, borehole, or vault alternative as shown in Table 5.3.11-1 of the *Draft GTCC EIS* (DOE 2011d).

[h] Includes waste from construction and operations. See Table 4-23.

[i] The LLW volume includes about 15 cubic meters of LLW in demolition debris that may also contain polychlorinated biphenyls in paint. This waste may be disposed of as LLW in E-Area (DOE 2014e).

Note: Total may not equal the sum of the contributions due to rounding. To convert cubic meters to cubic feet, multiply by 35.314; metric tons to tons, multiply by 1.1023.

Increases in the generation of solid LLW, hazardous waste, and solid nonhazardous waste are projected. LLW would be sent to E-Area for disposal or transported off site to DOE or commercial disposal facilities. Solid nonhazardous waste would continue to be disposed of at the Three Rivers Regional Landfill or an onsite construction and demolition landfill. Efforts would be made to recycle as much of the solid nonhazardous waste as reasonably possible to reduce the need for its disposal.

Although operation of the proposed biomass cogeneration and heating plants at D-, K-, and L-Areas would generate wood ash that would be disposed of at landfills such as the Three Rivers Regional Landfill, DOE expects an overall decrease in the quantities of solid nonhazardous wastes requiring disposal. This is because the biomass fuels to be burned in the new plants would reduce the amount of fly and bottom ash (compared to coal ash) entering SRS landfills by more than 95 percent. Furthermore, the biomass fuels to be burned would otherwise require disposal space in landfills (DOE 2008b).

Construction of Vogtle Units 3 and 4 would result in negligible quantities of solid hazardous and nonhazardous waste, whereas their operation would principally generate solid LLW and spent nuclear fuel. Generation of solid LLW is not expected to exceed 162 cubic meters (212 cubic yards) per year. Spent nuclear fuel would be stored on site until an offsite facility becomes available to accept this fuel. Some wastes generated at SRS and Vogtle could be disposed of at the same commercial facilities that could be used to dispose waste generated by the proposed activities. Wastes disposed of at the same commercial facilities would be within the permitted capacity and waste acceptance criteria for the facilities and therefore would have no incremental cumulative impacts.

Under the H-Area Alternative, vitrified HLW canisters would be generated at DWPF from the liquid HLW generated at H-Canyon, while under the L-Area Alternative, MCOs would be generated from a melt and dilute process. Under the H-Area Alternative, 15 to 101 additional HLW canisters could be produced at DWPF, while under the L-Area Alternative, 82 MCOs would be generated. Under the alternatives evaluated in the *Final SPD Supplemental EIS*, 5 to 100 additional HLW canisters could be generated (DOE 2015a). The maximum of 201 additional canisters would represent about 2 percent of the projected number of HLW canisters (8,582) in the current SRS Liquid Waste System Plan (SRR 2014b). DOE would store the vitrified HLW canisters in S-Area and the MCOs in L-Area at SRS pending offsite disposition.

Under the action alternatives evaluated in this Draft EA, there would be increases in the disposal of liquid radioactive waste at the saltstone facilities; the additional saltstone volume would range from 3.7 million to 6.2 million liters (0.98 million to 1.6 million gallons). This additional volume would represent a small fraction of the 662 million liters (175 million gallons) of waste expected to require treatment and disposal at the saltstone facilities (DOE 2001a).

4.3.2.7 Cumulative Impacts on Transportation

The assessment of cumulative impacts for transportation concentrates on radiological impacts from offsite transportation throughout the nation that would result in potential radiation exposure to the transportation crew and general population, in addition to those impacts evaluated in this EA. Cumulative radiological impacts from transportation are measured using the collective dose to the general population and workers because dose can be directly related to LCFs using a cancer risk coefficient.

The cumulative impacts from transport of radioactive material consist of impacts from historical shipments of radioactive waste and spent nuclear fuel; reasonably foreseeable actions that include transportation of radioactive material identified in Federal, non-Federal, and private environmental impact analyses; and general radioactive material transportation that is not related to a particular action. The timeframe of impacts was assumed to begin in 1943 and continue to some foreseeable future date. Projections for commercial radioactive material transport extend to 2073 based on available information.

The impacts from transportation in this EA are quite small compared with overall cumulative transportation impacts. The collective worker dose from all types of shipments is estimated to be about 421,000 person-rem (253 LCFs) for the period from 1943 through 2073 (131 years). The general population collective dose is estimated to be about 437,000 person-rem (262 LCFs). Worker and general population collective doses as estimated in this EA range from 0 to

10.9 person-rem and from 0 to 4.7 person-rem, respectively with no LCFs expected. To put these numbers in perspective, the National Center for Health Statistics indicates that the annual average number of cancer deaths in the United States from 1999 through 2004 was about 554,000, with less than a 1 percent fluctuation in the number of deaths in any given year (CDC 2008, 2011, 2012a, 2012b, 2013). The total number of LCFs (among the workers and the general population) estimated to result from radioactive material transportation over the period between 1943 and 2073 is 515, or an average of about 4 LCFs per year. The transportation-related LCFs represent about 0.0007 percent of the overall annual number of cancer deaths; indistinguishable from the national fluctuation in the total annual death rate from cancer. Note that the majority of the cumulative risks to workers and the general population would be due to the general transportation of radioactive material unrelated to activities evaluated in this EA.

4.3.2.8 Cumulative Impacts on Environmental Justice

Cumulative environmental justice impacts occur when the net effect of regional projects or activities results in disproportionately high and adverse human health and environmental effects on minority or low-income populations. The environmental justice analysis for alternatives in this EA indicates no high and adverse human health and environmental impacts on any population within the SRS ROI. Impacts on minority or low-income populations would be comparable to those on the population as a whole. Little to no change in radiological exposure is expected to occur during facility operations under all alternatives and options and therefore would not cumulatively contribute to environmental justice impacts. Therefore, no cumulative disproportionately high and adverse human health and environmental effects on minority or low-income populations are expected.

4.4 IRREVERSIBLE AND IRRETRIEVABLE COMMITMENT OF RESOURCES FOR THE PROPOSED ACTION

A commitment of resources is irreversible when primary or secondary impacts limit future options for a resource. A commitment of resources is irretrievable when resources that are used or consumed are neither renewable nor recoverable for future use. This section discusses the commitment of resources in four major categories: land, labor, utilities, and materials.

Activities occurring in the global commons and at Joint Base Charleston – Weapons Station would be of relatively short duration and would use a limited amount of non-renewable resources such as fuel for transport vehicles and heavy equipment. Therefore, these activities would be expected to result in minor irreversible and irretrievable commitment of resources.

Table 4-34 lists the commitments of resources related to construction activities at SRS for the various alternatives. Construction would require land, labor, utilities, and materials. For construction at SRS, there would be no change in land use at H- or L-Area and minimal land disturbance, and relatively minor commitments of labor, utilities, and materials.

Table 4-34: Commitments of Construction Resources at the Savannah River Site

Resource	Alternative			
	H-Area			L-Area
	Vitrification Option	LEU Waste Option	LEU/Thorium Waste Option	Melt and Dilute Option
Land Use				
Disturbed land (acres)	0.4	5.4	5.4	2.7
Labor				
Full-time equivalent (person-years)	270	420	420	430
Utilities				
Electricity (kilowatt-hours)	200	700	700	200
Diesel fuel (gallons)	11,000	26,000	26,000	11,000
Gasoline (gallons)	14,000	29,000	29,000	14,000
Water (gallons)	380,000	1,100,000	1,100,000	380,000
Materials				
Asphalt (cubic yards)	0	170	170	Minimal
Concrete (cubic yards)	0	6,000	6,000	550
Crushed stone, sand, and gravel (tons)	950	1,100	1,100	5,000
Lumber (square feet)	10,000	22,000	22,000	20,000
Soil (cubic yards)	0	5,000	5,000	Minimal
Steel (tons)	170	1,000	1,000	150

Note: To convert acres to hectares, multiply by 0.40469; gallons to liters, multiply by 3.7854; cubic yards to cubic meters, multiply by 0.76456; tons to metric tons, multiply by 0.90718; square feet to square meters, multiply by 0.092903; 1 full-time equivalent = 2,080 worker hours.
Source: DOE 2014a

Table 4-35 lists the commitments of resources related to operation activities at SRS for the H-and L-Area Alternatives. Operations would use labor, utilities and materials. Because large quantities of resources would not be used and the resources listed in Table 4-35 are not known to be in short supply, notable impacts from the irreversible and irretrievable commitment of resources are not expected under any of the alternatives. Note that some resources, such as water and steel, may be recycled after use and therefore are not truly irreversible or irretrievable.

Table 4-35: Commitments of Operations Resources at the Savannah River Site

Resource	Alternative			
	H-Area[a,b]			L-Area[b]
	Vitrification Option	LEU Waste Option	LEU/Thorium Waste Option	Melt and Dilute Option
Labor				
Full-time equivalent (person-years)	670	420	420	970
Utilities				
Electricity (megawatt-hours)	27,000	23,000	23,000	15,000
Steam (thousand lbs/yr)	47,000	57,000	57,000	18,000
Diesel fuel (gal/yr)	13,000	14,000	14,000	11,000
Water (gal/yr)	72,000,000	89,000,000	89,000,000	37,000,000
Materials				
Aluminum (kg/yr)	0	0	0	13,000
Aluminum nitrate (kg/yr)	5,200	5,200	5,200	0
Argon (l/yr)	0	500,000	500,000	0
Boric acid (kg/yr)	77	23	23	0
Calcium or Magnesium (kg/yr)	0	0	0	2,800
Copper formate (kg/yr)	650	200	200	0
Fly ash (kg/yr)	0	10,000	10,000	0
Formic acid (kg/yr)	25,000	7,600	7,600	0
Glass frit (kg/yr)	260,000	78,000	78,000	0
Hydrogen peroxide (kg/yr)	8,000	8,000	8,000	8,000
Nitric acid (kg/yr)	330,000	290,000	290,000	210,000
Nitrogen (l/yr)	0	2,000	2,000	0
Oxalic acid (kg/yr)	66,000	20,000	20,000	58
Portland cement (kg/yr)	0	30,000	30,000	0
Potassium fluoride (kg/yr)	500	500	500	0
Potassium nitrate (kg/yr)	77	23	23	0
Saltstone premix (kg/yr)	4,800,000	4,800,000	4,800,000	2,600,000
Slag (kg/yr)	14,000,000	4,100,000	4,100,000	0
Sodium hydroxide (kg/yr)	680,000	280,000	280,000	52,000
Sodium nitrate (kg/yr)	190,000	140,000	140,000	120,000
Sodium titanate (kg/yr)	94,000	28,000	28,000	0
Sodium tetraphenylborate (kg/yr)	5,800	1,700	1,700	0
Stainless Steel 304L (kg/yr)	26,000	46,000	46,000	14,000
Uranium, depleted (metric tons)	0	3.2[c]	3.2[c]	3.2
Zeolite, monosodium titanate, crystalline silicotitanate (kg/yr)	190	190	190	300
Zirconium oxide (kg/yr)	0	10,000	10,000	0

Note: To convert kilograms to pounds, multiply by 2.2046; liters to gallons, multiply by 0.26418; cubic feet to cubic meters, multiply by 0.028317; metric tons to tons, multiply by 1.1023; 1 full-time equivalent = 2,080 worker hours.

lbs = pounds; gal = gallon; yr = year, kg = kilogram; l = liter;

[a] Information related to operation of the Defense Waste Processing Facility was obtained from the *Final Supplemental Environmental Impact Statement, Defense Waste Processing Facility* (DOE 1994b).

[b] Information related to operation of the Salt Waste Processing Facility and Saltstone Production and Disposal Facilities was obtained from the *Savannah River Site, Salt Processing Alternatives Final Supplemental Environmental Impact Statement* (DOE 2001a).

[c] 3.2 metric tons delivered as 2,100 gallons of uranyl nitrate or 3,850 kilograms of uranium trioxide powder

Source: DOE 1994b, 2001a, and 2014a

4.5 RELATIONSHIP BETWEEN LOCAL SHORT-TERM USES OF THE ENVIRONMENT AND MAINTENANCE AND ENHANCEMENT OF LONG-TERM PRODUCTIVITY FOR THE PROPOSED ACTION

Activities occurring in the global commons and at Joint Base Charleston – Weapons Station would be of relatively short duration and would be conducted in a manner similar to ongoing activities. Therefore, these short-term uses of the environment would not be expected to result in an incremental change in the potential long-term productivity of these sites.

The relationship between short-term uses of the environment and the maintenance and enhancement of long-term productivity for key environmental resources at SRS is described in the following paragraphs:

- Small areas of land would be disturbed in H- and L-Areas to construct or modify new or existing facilities. The construction activities would be within developed industrial landscapes at H- and L-Areas. After the operational life of the facilities, DOE could deactivate, decontaminate, and decommission the facilities in accordance with applicable regulatory requirements and then close in place or restore the areas occupied by the facilities to brownfield sites that would be available for other industrial uses. Appropriate CERCLA and/or NEPA reviews would be conducted before initiation of decontamination and decommissioning actions. In all likelihood, none of the sites would be restored to a natural terrestrial habitat.
- Groundwater would be used to meet process and sanitary water needs over the duration of the project. After use, most of this water would be treated and released through permitted outfalls into surface water streams. The withdrawal, use, and treatment of water are not likely to affect the long-term productivity of this resource.
- Air emissions associated with implementation of any of the alternatives would add small amounts of radiological and nonradiological constituents to the air of the SRS region. These emissions would result in additional radioactive exposure or air loading, but are not expected to affect compliance by SRS with radiation exposure or air quality standards. No substantial residual environmental effects on long-term environmental productivity are expected.
- The management and disposal of LLW and solid and liquid wastes would require energy and space at treatment, storage, and disposal facilities at SRS (e.g., Z-Area saltstone facilities, E-Area Vaults, Three Rivers Regional Landfill). Areas used at SRS for LLW and solid waste disposal would require a long-term commitment of land resources.

The offsite management and disposal of HLW and LLW would require energy and space at the treatment, storage, and disposal facilities. Areas used for HLW and LLW disposal would require a long-term commitment of land resources.

4.6 MITIGATION MEASURES FOR THE PROPOSED ACTION

As specified in the Council on Environmental Quality's NEPA regulations (40 CFR 1508.20), mitigation includes: avoiding the impact altogether by not taking a certain action or parts of an action; minimizing impacts by limiting the degree or magnitude of the action and its implementation; rectifying the impact by repairing, rehabilitating, or restoring the affected environment; reducing or eliminating the impact over time by preservation and maintenance operations during the life of the action; and compensating for the impact by replacing or providing substitute resources or environments.

In general, activities associated with construction and operation of facilities would follow standard practices such as Best Management Practices (BMPs) for minimizing impacts on environmental resources as required by regulation, permit, or guidelines. For any alternative, stewardship practices that are protective of the air, water, land, and other natural and cultural resources affected by DOE operations would be implemented in accordance with an environmental management system established pursuant to DOE Order 436.1, *Departmental Sustainabilit*y, which was prepared to incorporate the requirements of Executive Order 13514, *Federal Leadership in Environmental, Energy, and Economic Performance*.[39]

As described earlier in this chapter, because no substantial adverse impacts are expected, no mitigation measures beyond those required by regulation or achieved through BMPs would be needed.

[39] Section 16 of EO 13693, *Planning for Federal Sustainability in the Next Decade*, revokes Executive Order 13514.

5 GLOSSARY

aquifer—A body of rock or sediment that is capable of transmitting groundwater and yielding usable quantities of water to wells or springs.

aquitard—A less-permeable, or impermeable, geologic unit in a stratigraphic sequence. Aquitards separate aquifers.

as low as reasonably achievable (ALARA)—An approach to radiation protection to manage and control worker and public exposures (both individual and collective) and releases of radioactive material to the environment to as far below applicable limits as social, technical, economic, practical, and public policy considerations permit. ALARA is not a dose limit, but a process for minimizing doses to as far below limits as is practicable.

background radiation—Radiation from (1) cosmic sources; (2) naturally occurring radioactive materials, including radon (except as a decay product of source or special nuclear material); and (3) global fallout as it exists in the environment (e.g., from the testing of nuclear explosive devices).

beyond-design-basis accident—This term is used as a technical way to discuss accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely. (In that sense, they are considered beyond the scope of design-basis accidents [e.g., fire, earthquake, spill, explosion] that a nuclear facility must be designed and built to withstand.) As the regulatory process strives to be as thorough as possible, "beyond-design-basis" accident sequences are analyzed to fully understand the capability of a design. These accidents are typically very low-probability, but high-consequence events. (See design-basis accident.)

Carolina bay—Closed, elliptical depressions capable of holding water, common on and near SRS. A Carolina bay is generally considered a type of wetland.

criticality—The condition in which a system undergoes a sustained nuclear chain reaction.

decay (radioactive)—The decrease in the amount of any radioactive material with the passage of time, due to spontaneous nuclear disintegration (i.e., emission from atomic nuclei of charged particles, photons, or both).

depleted uranium—Uranium with a content of the fissile isotope uranium-235 of less than 0.7 percent (by weight) found in natural uranium, so that it contains more uranium-238 than natural uranium.

design-basis—For nuclear facilities, information that identifies the specific functions to be performed by a structure, system, or component and the specific values (or ranges of values) chosen for controlling parameters for reference bounds for design. These values may be (1) restraints derived from generally accepted, state-of-the-art practices for achieving functional goals; (2) requirements derived from analysis (based on calculation or experiment) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals; or (3) requirements derived from Federal safety objectives, principles, goals, or requirements.

design-basis accident—An accident postulated for the purpose of establishing functional and performance requirements for safety structures, systems, and components. (See beyond-design-basis accident.)

documented safety analysis (DSA)—A report that systematically identifies potential hazards within a nuclear facility, describes and analyzes the adequacy of measures to eliminate or control identified hazards, and analyzes potential accidents and their associated risks. Safety analysis reports are used to ensure that a nuclear facility can be constructed, operated, maintained, shut down, and decommissioned safely and in compliance with applicable laws and regulations. Safety analysis reports (or documented safety analyses per 10 CFR Part 830) are required for U.S. Department of Energy (DOE) nuclear facilities and as a part of applications for U.S. Nuclear Regulatory Commission (NRC) licenses. The NRC regulations or DOE orders and technical standards that apply to the facility type provide specific requirements for the content of safety analysis reports. (See nuclear facility.)

dose—A generic term meaning absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, or committed equivalent dose. For ionizing radiation, the energy imparted to matter by ionizing radiation per unit mass of the irradiated material (e.g., biological tissue). The units of absorbed dose are the rad and the gray. In many publications, the rem is used as an approximation of the rad.

effective dose equivalent—The dose value obtained by multiplying the dose equivalents received by specified tissues or organs of the body by the appropriate weighting factors applicable to the tissues or organs irradiated, and then summing all of the resulting products. It includes the dose from radiation sources internal and external to the body. The effective dose equivalent is expressed in units of rem or sieverts.

enriched uranium—Uranium whose content of the fissile isotope uranium-235 is greater than the 0.7 percent (by weight) found in natural uranium. (See highly enriched uranium and low-enriched uranium.)

environmental assessment (EA)—A concise public document that a Federal agency prepares under the National Environmental Policy Act (NEPA) to provide sufficient evidence and analysis to determine whether a proposed agency action would require preparation of an environmental impact statement (EIS) or a finding of no significant impact. A Federal agency may also prepare an EA to aid its compliance with NEPA when no EIS is necessary or to facilitate preparation of an EIS when one is necessary.

environmental justice—The fair treatment and meaningful involvement of all people regardless of race, color, national origin, or income with respect to the development, implementation, and enforcement of environmental laws, regulations, and policies. Fair treatment means that no group of people, including racial, ethnic, or socioeconomic groups, should bear a disproportionate share of the negative environmental consequences resulting from industrial, municipal, and commercial operations or the execution of Federal, state, local, and tribal programs and policies. Executive Order 12898 directs Federal agencies to make achieving environmental justice part of their missions by identifying and addressing disproportionately high and adverse effects of agency programs, policies, and activities on minority and low-income populations. (See minority population and low-income population.)

Finding of No Significant Impact (FONSI)—A public document issued by a Federal agency briefly presenting the reasons why an action for which the agency has prepared an environmental assessment has no potential to have a significant effect on the human environment and, thus, will not require preparation of an environmental impact statement. (See environmental assessment and environmental impact statement.)

fissile material—Although sometimes used as a synonym for fissionable material, this term has acquired a more restricted meaning; namely, any material fissionable by low-energy (i.e., thermal or slow) neutrons. Fissile materials include uranium-233 and -235, and plutonium-239 and -241.

fission—A nuclear transformation that is typically characterized by the splitting of the nucleus of a heavy nucleus into at least two other nuclei, the emission of one or more neutrons, and the release of a relatively large amount of energy. Fission of heavy nuclei can occur spontaneously or be induced by neutron bombardment.

fission products—Nuclei (i.e., fission fragments) formed by the fission of heavy elements, in addition to the nuclides formed by the fission fragments' radioactive decay.

fugitive emissions—(1) Emissions that do not pass through a stack, vent, chimney, or similar opening where they could be captured by a control device, or (2) any air pollutant emitted to the atmosphere other than from a stack. Sources of fugitive emissions include pumps; valves; flanges; seals; area sources such as ponds, lagoons, landfills, and piles of stored material (such as coal); and road construction areas or other areas where earthwork is occurring.

half-life (radiological)—Time in which one-half of the atoms of a particular radionuclide disintegrate into another nuclear form. Half-lives for specific radionuclides vary from millionths of a second to billions of years.

hazardous material—A material, as defined by 49 CFR 171.8, that the Department of Transportation has determined is capable of posing an unreasonable risk to health, safety, and property when transported in commerce.

hazardous air pollutants—Air pollutants not covered by ambient air quality standards, but that may present a threat of adverse human health or environmental effects. Those specifically listed in 40 CFR 61.01 are asbestos, benzene, beryllium, coke oven emissions, inorganic arsenic, mercury, radionuclides, and vinyl chloride. More broadly, hazardous air pollutants are any of the 189 pollutants listed in or pursuant to Section 112(b) of the Clean Air Act.

high-level radioactive waste (HLW)—As defined in the Nuclear Waste Policy Act of 1982, as amended, means (A) the highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (B) other highly radioactive material that the U.S. Nuclear Regulatory Commission (NRC), consistent with existing law, determines by rule requires permanent isolation.

highly enriched uranium (HEU)—Uranium whose content of the fissile isotope uranium-235 has been increased through enrichment to 20 percent or more (by weight). Highly enriched uranium can be used in making nuclear weapons and also as fuel for some isotope-production, research, naval propulsion, and power reactors. (See enriched uranium and low-enriched uranium.)

ion exchange—A physiochemical process that removes anions and cations, including radionuclides, from liquid streams (usually water) for the purpose of purification or decontamination.

ionizing radiation—Particles (alpha, beta, neutrons, and other subatomic particles) or photons (i.e., gamma, x-rays) emitted from the nucleus of unstable atoms as a result of radioactive decay. Such radiation is capable of displacing electrons from atoms or molecules in the target material (such as biological tissues), thereby producing ions.

isotope—Any of two or more variations of an element in which the nuclei have the same number of protons (and thus the same atomic number), but different numbers of neutrons so that their atomic masses differ. Isotopes of a single element possess almost identical chemical properties, but often different physical properties; e.g., carbon-12 and -13 are stable; carbon-14 is radioactive.

job control waste—Plastic sheeting, paper, small pieces of wood and metal, glass, gloves, protective clothing, and/or pieces of small equipment that were used in a radioactive process.

low-enriched uranium (LEU)—Uranium whose content of the fissile isotope uranium-235 has been increased through enrichment to more than 0.7 percent but less than 20 percent by weight. Most nuclear power reactor fuel contains low-enriched uranium containing 3 to 5 percent uranium-235. (See enriched uranium and highly enriched uranium.)

low-level radioactive waste (LLW)—Radioactive waste that is not high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in Section 11e.(2), (3), or (4) of the Atomic Energy Act of 1954, as amended.

material at risk (MAR)—The amount of radionuclides in curies of activity or grams for each radionuclide available for release when acted upon by a given physical insult, stress, or accident. The material at risk is specific to a given process in the facility of interest. It is not necessarily the total quantity of material present, but it is that amount of material in the scenario of interest postulated to be available for release.

maximally exposed individual (MEI)—A hypothetical individual whose location and habits result in the highest total radiological or chemical exposure (and thus dose) from a particular source for all exposure routes (i.e., inhalation, ingestion, direct exposure, resuspension).

natural phenomena hazard—A category of events (e.g., earthquake, severe wind, tornado, flood, and lightning) that must be considered in the U.S. Department of Energy facility design, construction, and operations, as specified in DOE Order 420.1B.

nonproliferation—Preventing the spread of nuclear weapons, nuclear weapons materials, or nuclear weapons technology to rogue nations, terrorists, and countries that have not signed nonproliferation agreements.

nuclear criticality—See criticality.

nuclear facility—A facility that is subject to requirements intended to control potential nuclear hazards. Defined in U.S. Department of Energy directives as any nuclear reactor or any other facility whose operations involve radioactive materials in such form and quantity that a significant nuclear hazard potentially exists to the employees and/or the general public.

person-rem—A unit of collective radiation dose applied to populations or groups of individuals; that is, a unit for expressing the dose when summed across all persons in a specified population or group. One person-rem equals 0.01 person-sieverts.

proliferation—The spread of nuclear, biological, or chemical capabilities and the weapons (i.e., missiles) capable of delivering them.

rad—A unit of radiation-absorbed dose (e.g., in body tissue). One rad is equal to an absorbed dose of 0.01 joules per kilogram.

radiation—See ionizing radiation.

radioactivity— Defined as a *process*: The spontaneous transformation of unstable atomic nuclei, usually accompanied by the emission of ionizing radiation.

Defined as a *property*: The property of unstable nuclei in certain atoms to spontaneously emit ionizing radiation during nuclear transformations.

radionuclide—A radioactive element characterized according to its atomic mass and atomic number. Radionuclides can be manmade or naturally occurring, have a long half-life, and have potentially mutagenic, teratogenic, or carcinogenic effects on the human body.

radon—A colorless, odorless, naturally occurring, radioactive, inert, gaseous element formed by radioactive decay of radium atoms. The atomic number is 86.

region of influence (ROI)—The physical area that bounds the environmental, sociological, economic, or cultural features of interest for the purpose of analysis.

rem—See roentgen equivalent man.

remote-handled waste—In general, refers to radioactive waste that must be handled at a distance to protect workers from unnecessary exposure.

repository—A facility for disposal of radioactive waste.

roentgen—A unit of exposure to ionizing x-ray or gamma radiation equal to or producing 1 electrostatic unit of charge per cubic centimeter of air. It is approximately equal to 1 rad.

roentgen equivalent man (rem)—A unit of dose equivalent. The dose equivalent in rem equals the absorbed dose in rad in tissue multiplied by the appropriate quality factor and possibly other modifying factors. Rem refers to the dosage of ionizing radiation that will cause the same biological effect as one roentgen of x-ray or gamma ray exposure. One rem equals 0.01 sieverts.

security—An integrated system of activities, systems, programs, facilities, and policies for the protection of Restricted Data and other classified information or matter, nuclear materials, nuclear weapons and nuclear weapons components, and/or U.S. Department of Energy or contractor facilities, property, and equipment.

shielding—Any material or obstruction (e.g., bulkhead, wall, or other structure) that absorbs radiation, and thus tends to protect personnel or materials from the effects of ionizing radiation.

special nuclear material—As defined in the Atomic Energy Act of 1954, as amended, means 1) plutonium, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the Commission, pursuant to provisions of 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— As defined in the Nuclear Waste Policy Act of 1982, as amended, fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing.

stabilize—To convert a compound, mixture, or solution to a nonreactive form.

transuranic (TRU) element—Of, relating to, or being any radioactive element whose atomic number is higher than that of uranium (i.e. atomic number 92), including neptunium, plutonium, americium, and curium.

transuranic waste—Waste containing more than 100 nanocuries (3,700 becquerels) of alpha-emitting transuranic isotopes per gram of waste, with half-lives greater than 20 years, except for (A) high-level radioactive waste; (B) waste that the U.S. Department of Energy has determined, with the concurrence of the U.S. Environmental Protection Agency, does not need the degree of isolation called for by 40 CFR Part 191; or (C) waste that the U.S. Nuclear Regulatory Commission has approved for disposal on a case-by-case in accordance with 10 CFR Part 61.

uranium—A radioactive, metallic element with the atomic number 92. Uranium has 14 known isotopes, of which uranium-238 is the most abundant in nature. Uranium-235 is commonly used as a fuel for nuclear fission, and uranium-238 is transformed into fissionable plutonium-239 following its capture of a neutron in a nuclear reactor.

vitrification—A process by which finely ground glass (e.g., borosilicate glass) is used to immobilize radioactive wastes.

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Appendix A: Statement of Intent

Statement of Intent

between

the Federal Ministry of Education and Research of the Federal Republic of Germany

and

the Ministry for Innovation, Science and Research of the State of North Rhine-Westphalia on behalf of the North Rhine-Westphalian State Government

the Department of Energy of the United States of America

for the Proposed Use of Savannah River Site Facilities for Disposition of German Research Reactor Pebble Bed Fuel

I. Background

1. The Department of Energy (DOE) of the United States of America, in cooperation with the Federal Ministry of Education and Research (BMBF) of the Federal Republic of Germany and the Ministry for Innovation, Science and Research of the State of North Rhine-Westphalia on behalf of the North Rhine-Westphalian State Government (hereinafter collectively "the Participants"), is considering the feasibility of DOE acceptance of graphite-based spent nuclear fuel that contains United States-origin highly enriched uranium (HEU) and has been determined to have been irradiated in Germany for research and development purposes (hereinafter "the German Research Reactor Pebble Bed Fuel" or "the fuel"), and disposition of the fuel using DOE facilities at the Savannah River Site (SRS), near Aiken, South Carolina.

2. DOE's acceptance of the fuel would support the United States' HEU minimization policy objective of seeking to reduce, and eventually to eliminate, HEU from civil commerce by removing United States-origin HEU from Germany and returning it to the United States for safe storage and disposition, and converting it into a form no longer usable for a nuclear weapon or an improvised nuclear material dispersal device. Disposition of the fuel would also contribute to the objectives of the Nuclear Security Summit in 2014.

3. The fuel under consideration is coated HEU/thorium fuel kernels embedded in a spherical graphite matrix used in the early research and development of pebble bed reactors.

4. DOE is considering the feasibility of using H-Canyon facilities at SRS to chemically remove the graphite from the fuel kernels. Based on positive results of research and development done to date, it appears technically feasible to utilize the H-Canyon facilities at SRS to chemically remove the graphite from the fuel kernels by using a molten salt technique being developed by the Savannah River National Laboratory. The remaining fuel kernels could then be processed through the H-Canyon system for disposition.

5. In consideration of the foregoing, the Participants express their willingness to engage in cooperation with the aim of creating the necessary prerequisites no later than the first quarter of 2015 for conclusion of an appropriate legal framework for returning the fuel to the United States.

II. Planned Cooperation

The Participants intend to take the following actions as soon as possible.

1. DOE (or its contractors) is/are to conduct any reviews and work required by United States law for acceptance of the fuel and its processing and disposition. This includes compliance with all applicable requirements of the United States National Environmental Policy Act (NEPA). The Participants share the conviction that a NEPA review should start as soon as possible.

2. The NEPA review is to include preparation of an environmental assessment (EA) to analyze the potential environmental consequences of the proposed acceptance, processing, and disposition of the fuel, and providing notice of DOE's intent to prepare an EA to the States of Georgia and South Carolina, and the general public, and publication of such notice in the United States' *Federal Register*. The EA is to inform DOE's decision whether to issue a finding of no significant impact, which would conclude the NEPA review, or prepare a more detailed environmental impact statement.

3. DOE intends to undertake other activities mutually decided by the Participants, including additional technical and engineering work, and project management, in order for DOE to reach a decision on the proposed acceptance, processing, and disposition of the German Research Reactor Pebble Bed Fuel. The NEPA review and the activities necessary to support NEPA described in this paragraph constitute the preparatory phase.

4. *Forschungszentrum Jülich* (FZJ) is to bear the costs of the preparatory phase work and, if there is a decision to proceed with the project, the costs associated with the acceptance, processing, and disposition of the fuel. DOE and FZJ should finalize as soon as possible a contract for completion of the preparatory phase. If there is a decision to proceed with the project, the terms and conditions for acceptance, processing, and disposition of the fuel should be set forth in a contract, including provisions to be applied in the event of premature termination of the project.

III. General Considerations

1. Cooperative activities under this Statement of Intent are to commence immediately after signature by the Participants.

2. Each Participant is to conduct the activities contemplated by this Statement of Intent in accordance with all applicable laws and regulations and any international agreements to which its government is a party.

3. Cooperative activities under this Statement of Intent are subject to the availability of funds, personnel, and other resources.

4. This Statement of Intent does not create any legally binding obligations between or among the Participants.

5. This Statement of Intent may be revised at any time in writing by the Participants' mutual consent in writing.

6. The Participants may discontinue this Statement of Intent at any time. A Participant that decides to discontinue its participation in the activities under this Statement of Intent should provide prompt advance notice in writing to the other Participants.

7. Any decision by the Participants to proceed with the transportation of the fuel for storage, processing, and disposition depends upon compliance with all applicable requirements of United States law and DOE requirements, including NEPA, and resolution by the Participants of any technical, financial, and legal issues that may be identified during consideration of the feasibility of the project and development of an appropriate legal framework.

Signed in three originals, at Washington on the 28th day of March 2014 and at Bonn on the 01. day of April 2014.

For the Federal Ministry of Education
and Research of the Federal Republic
of Germany:

W. G

For the Ministry for Innovation, Science and Research
of the State of North Rhine-Westphalia
on behalf of the North Rhine-Westphalian State Government:

For the Department of Energy
of the United States of America: