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

Volume 1
Appendix C

Savannah River Site Spent Nuclear Fuel Management Program

June 1994

U.S. Department of Energy
Office of Environmental Management
Idaho Operations Office

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1. INTRODUCTION

The U.S. Department of Energy (DOE) is engaged in two related decisionmaking processes concerning: (1) the transportation, receipt, processing, and storage of spent nuclear fuel (SNF) at the DOE Idaho National Engineering Laboratory (INEL) which will focus on the next 10 years; and (2) programmatic decisions on future spent nuclear fuel management which will emphasize the next 40 years.

DOE is analyzing the environmental consequences of these spent nuclear fuel management actions in this two-volume Environmental Impact Statement (EIS). Volume 1 supports broad programmatic decisions that will have applicability across the DOE complex and describes in detail the purpose and need for this DOE action. Volume 2 is specific to actions at the INEL. This document, which limits its discussion to the Savannah River Site (SRS) spent nuclear fuel management program, supports Volume 1 of the EIS. Other documents supporting Volume 1 focus on spent nuclear fuel management programs for the Hanford Site, INEL, Naval Nuclear Propulsion Program, and other sites.

As part of its planning process for this two-volume EIS, DOE issued an Implementation Plan on October 29, 1993. The organization of this document is consistent with the provisions established in the Implementation Plan and are outlined below:

- Chapter 2 contains background information related to the SRS and the framework of environmental regulations pertinent to spent nuclear fuel management.

- Chapter 3 identifies spent nuclear fuel management alternatives that DOE could implement at the SRS, and summarizes their potential environmental consequences.

- Chapter 4 describes the existing environmental resources of the SRS that spent nuclear fuel activities could affect.

- Chapter 5 analyzes in detail the environmental consequences of each spent nuclear fuel management alternative and describes cumulative impacts. The chapter also contains information on unavoidable adverse impacts, commitment of resources, short-term use of the environment and mitigation measures.

2. BACKGROUND

This chapter contains an overview of the Savannah River Site (SRS) and a description of the regulatory framework related to the actions that this document evaluates. In addition, it discusses the U.S. Department of Energy (DOE) Spent Nuclear Fuel (SNF) Management Program as it relates to the SRS. Finally, it describes the representative sites located on the SRS that could serve as locations for spent nuclear fuel facilities.

2.1 SRS Overview

The SRS is a key DOE facility for research on and processing of special nuclear materials. The U.S. Government built the Site in the early 1950s to produce the basic materials - primarily plutonium-239 and tritium - used in the fabrication of nuclear weapons. The DOE Savannah River Operations Office manages the SRS, and Westinghouse Savannah River Company (WSRC) operates the Site under contract to DOE.

2.1.1 Site Description

The SRS occupies an area of approximately 310 square miles (800 square kilometers) in western South Carolina, in a generally rural area about 25 miles (40 kilometers) southeast of Augusta, Georgia, and 12 miles (19 kilometers) south of Aiken, South Carolina (Figure 2-1). The Savannah River forms the southwestern border of the SRS, which includes portions of Aiken, Barnwell, and Allendale Counties. The average population density (1990 census data) in the six-county region of influence around the Site is 140 people per square mile (54 per square kilometer); the largest concentration is 2,595 people per square mile (1,002 per square kilometer) in the City of Augusta (HNUS 1992). Four other population centers — Aiken, Allendale, Barnwell, and North Augusta, South Carolina — are within 22 miles (40 kilometers) of the Site. Three small towns — Jackson, New Ellenton, and Snelling, South Carolina — are adjacent to the SRS boundary to the northwest, north, and east, respectively. Based on 1990 U.S. Census Bureau data, the population within a 50-mile (80-kilometer) radius of the SRS is approximately 620,100 (Arnett et al. 1993).

The Site consists primarily of managed upland forest with some wetland areas. Facilities and roadways occupy approximately 5 percent of the SRS land area. Access to the Site is controlled, with

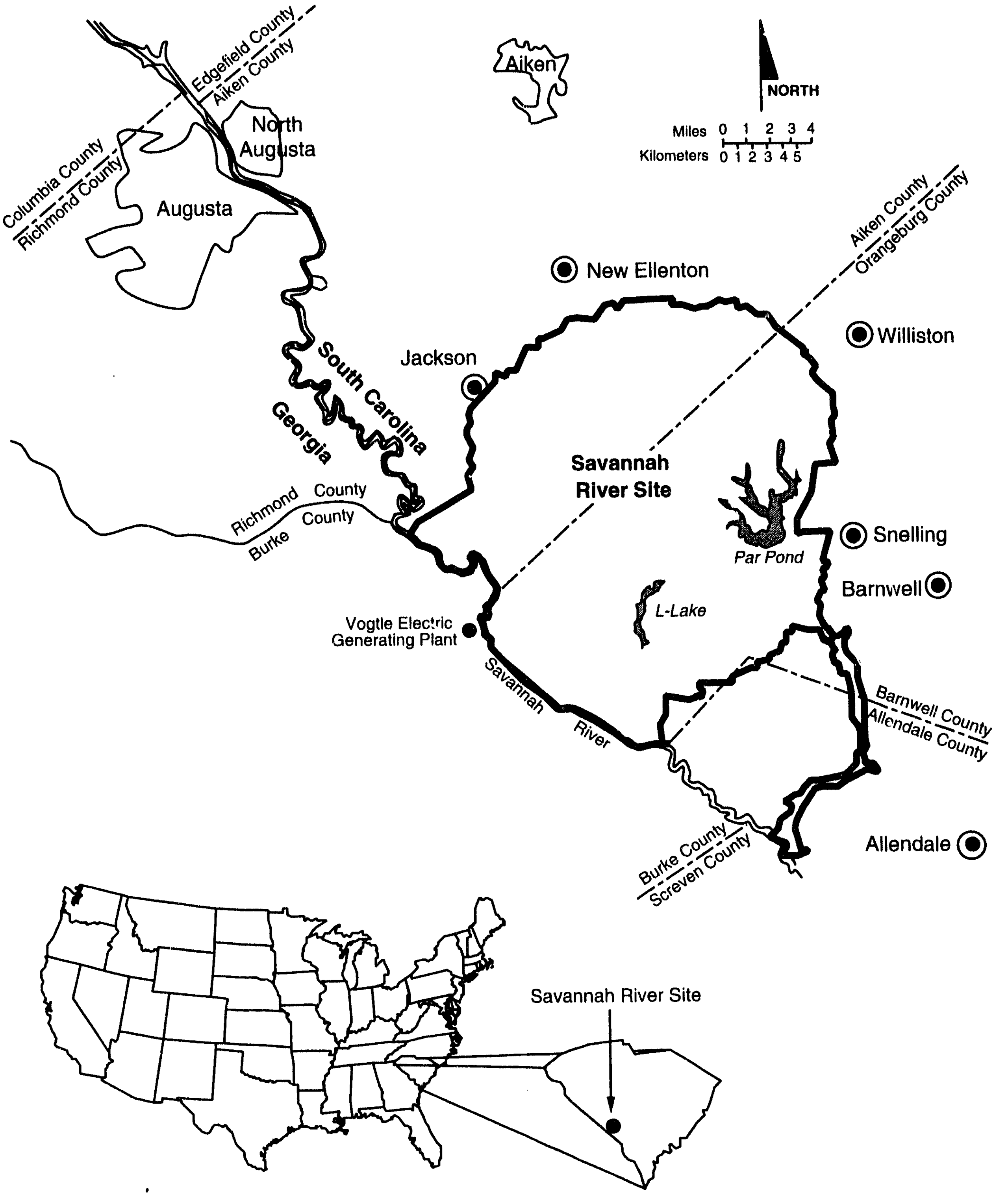

Figure 2-1. National location of SRS.

public transportation limited to through traffic on South Carolina Highway 125 (SRS Road A), U.S. Highway 278, SRS Road 1, and the CSX Railroad corridor.

The SRS contains 15 major production, service, and research and development (R&D) areas that previously supported nuclear materials production and can support processing operations and waste management activities. Major SRS facilities include five nuclear reactors, two chemical separations plants, a fuel and target fabrication facility, the Defense Waste Processing Facility (DWPF), the Replacement Tritium Facility, a heavy-water rework plant, and the Savannah River Technology Center (SRTC), formerly called the Savannah River Laboratory. In addition, the University of Georgia Research Foundation operates the Savannah River Ecology Laboratory (SREL) on the Site under contract to DOE. Under an interagency agreement, the U.S. Forest Service operates the Savannah River Forest Station, which manages the natural resources and secondary roads on the Site. These facilities are in defined areas scattered across the Site. Each area is identified by a letter designation, as summarized in Table 2-1. Figure 2-2 shows the locations of the principal SRS facilities. The reactor, waste storage, and separations areas are at least 4 miles (6 kilometers) inside the nearest SRS boundary.

The primary SRS facilities were related to the production of nuclear materials. M-Area manufactured fuel and target components for shipment to the SRS reactors. Originally, the Site operated five reactors; at present, all are in shutdown status. Shielded railroad cars transported irradiated fuel to the F- or H-Area Canyon for the recovery of nuclear materials. The F- and H-Area separations processes dissolve irradiated components in acid, and extract and separate the desired nuclear materials. In H-Area, additional processes extract other products from irradiated components.

DOE neutralizes and stores the high-level liquid radioactive waste generated by the separations facilities in underground tanks. DOE plans to process this waste into a borosilicate glass waste form in the Defense Waste Processing Facility when that facility becomes operational, and to store this glass waste form at the SRS until an offsite geological repository is available. [DOE is preparing a Supplemental EIS related to Defense Waste Processing Facility operations (59 FR 16499, 4/6/94).] In addition to the underground waste storage tanks, DOE has established a centrally located 196-acre (0.8-square-kilometer) site between F- and H-Areas, called E-Area, for the disposal of solid low-level radioactive waste and the storage of transuranic (TRU) radioactive waste and mixed (hazardous and radioactive) waste. The Site also has a central sanitary landfill and buildings in the Central Shops

Table 2-1. Description of functions and principal facilities at SRS areas.

Area	Function	Principal facilities
A	Main DOE administration area, research laboratories	Main administration building, Savannah River Technology Center, Savannah River Ecology Laboratory, powerhouse
B	Wackenhut Services, Inc., administration area (security)	Administration building, WSRC Engineering building, WSRC training buildings
C	One of five SRS reactors	C-Reactor, training facilities, cooling basin
D	Central powerhouse and heavy-water rework	Powerhouse, heavy-water rework facility
E	Waste disposal and storage	Solid Waste Disposal Facility
F	Process plutonium	F-Area Canyon, FB-Line, tank farm
G	Various support functions	Spread throughout the Site: railroad yard, U.S. Forest Service installations
H	Process uranium and tritium	H-Area Canyon, HB-Line, Effluent Treatment Facility, tank farm, Receiving Basin for Offsite Fuels, Consolidated Incineration Facility
K	One of five SRS reactors	K-Reactor, cooling basins, cooling tower
L	One of five SRS reactors	L-Reactor, cooling basins
M	Production of fuel and target assemblies	Slug and target production facilities, effluent treatment facility
N	Receiving	Central Shops
P	One of five SRS reactors	P-Reactor, cooling basins
R	One of five SRS reactors	R-Reactor, cooling basins
S	Process high-level radioactive waste	Defense Waste Processing Facility
TNX	Applied research and development	Analytical laboratory, Defense Waste Processing Technology facilities, various mockups, effluent treatment facilities
Z	Waste treatment and handling	Saltstone facility

(N Area) for the storage of nonradioactive hazardous wastes and mixed waste. DOE is preparing an EIS on waste management activities at the SRS (59 FR 16194; 4/6/94).

The Site contains facilities for processing support and for research and development. These include operational coal-fired powerhouses in A-, D-, and H-Areas that generate electricity and steam. The largest powerhouse, which is in D-Area, produces electricity and sends process steam to C-, F-,

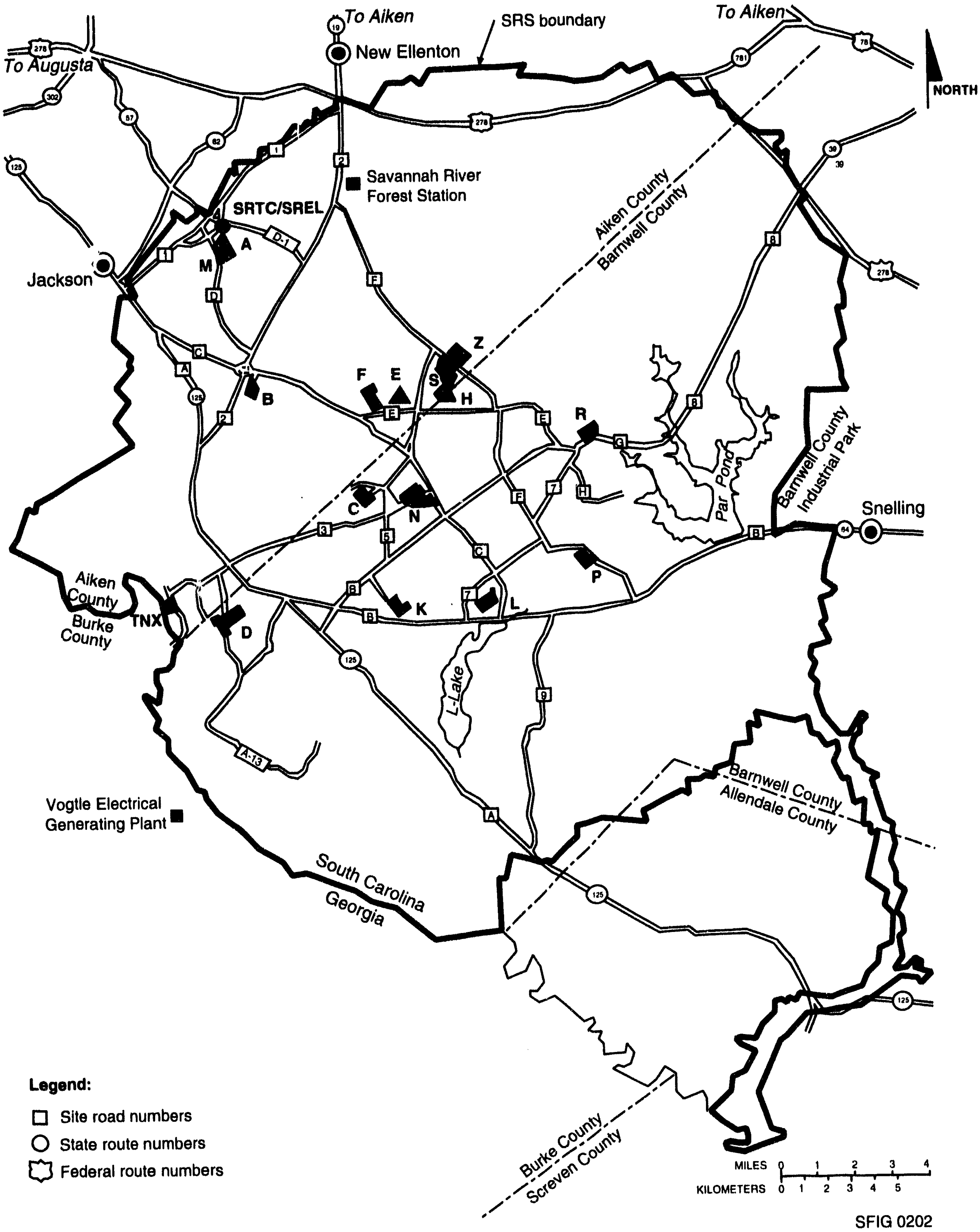

Figure 2-2. Location of principal SRS facilities (see Table 2-1).

H-, and S-Areas through a 7-mile (11-kilometer) steam line. D-Area also contains the heavy-water rework facility at which DOE purified the deuterium oxide (heavy water) used as the moderator and coolant in SRS reactors. TNX-Area facilities study chemical and waste processing problems and test production-scale equipment. Finally, A-Area facilities include the Savannah River Technology Center, the Savannah River Ecology Laboratory, and the DOE and Westinghouse Savannah River Company administrative offices.

The SRS employs approximately 21,000 people. Most of these employees work for Westinghouse Savannah River Company and its subcontractors. The remainder work for DOE, the Savannah River Ecology Laboratory, Wackenhut Services, Inc., the U.S. Forest Service, and other contractors.

2.1.2 Site History

The U.S. Atomic Energy Commission (AEC), a DOE predecessor agency, selected the location for the SRS in November 1950 after a study of more than 100 prospective sites. The government selected E. I. du Pont de Nemours and Company, Inc., to build and operate the facility. Construction began in February 1951; the basic plant was completed in 1956 at a cost of $1.1 billion, including the land. On October 3, 1952, operations began with the startup of a unit of the heavy-water extraction plant. Criticality occurred in the first production reactor on December 28, 1953.

In 1972, the AEC designated the SRS as the nation's first National Environmental Research Park. Through the years, scientists have performed a wide range of investigations on the diverse habitats, flora, and fauna of the Site.

2.1.3 Mission

The historic mission of the SRS was to serve the national security interests of the United States by safely processing nuclear materials while protecting the health and safety of employees and the public and protecting the environment. The SRS was responsible for producing tritium and special nuclear materials for national defense. At present, it supports the viability of the weapons stockpile by recycling limited-life components. The SRS also produces isotopes for nonweapons applications in the nation's space program and for medical applications.

The SRS spent nuclear fuel mission is to manage DOE-owned spent fuel in a cost-effective way that protects the safety of SRS workers, the public, and the environment. The focuses of near-term activities are the accurate quantification and characterization of DOE-owned spent nuclear fuel, assessment of spent nuclear fuel storage facilities, mitigation of current spent nuclear fuel storage vulnerabilities, and identification of technologies and requirements for interim management and ultimate disposition of spent nuclear fuel.

2.1.4 Management

The DOE Savannah River Operations Office manages the SRS; the Westinghouse Savannah River Company operates the Site under contract to DOE. Westinghouse assumed operational responsibility in April 1989 from E. I. du Pont de Nemours and Company, Inc., which had operated the Site since 1951.

2.2 Regulatory Framework

This section summarizes the framework of environmental protection regulations applicable to spent nuclear fuel management at the SRS. The framework is based on Federal and South Carolina laws and one local ordinance, as discussed below. Volume 1 (Section 7.3) of this Environmental Impact Statement (EIS) provides additional information on the major Federal environmental laws and regulations, Executive Orders, and DOE Orders that apply to spent nuclear fuel management alternatives.

2.2.1 Federal

The U.S. Environmental Protection Agency (EPA) has authorized South Carolina to implement most provisions of the Clean Air Act, Resource Conservation and Recovery Act, and Clean Water Act that apply to SRS spent nuclear fuel management. EPA Region IV has the lead responsibility for Clean Air Act standards for radionuclide emissions from DOE facilities, imposing monitoring and approval requirements on SRS spent nuclear fuel management activities that could result in radionuclide emissions.

In addition, EPA Region IV has Resource Conservation and Recovery Act authority over radioactive hazardous (mixed) waste management, affecting wastes from spent nuclear fuel processing.

EPA Region IV and the DOE Savannah River Operations Office have entered a Federal Facility Compliance Agreement on SRS mixed waste management.

The U.S. Army Corps of Engineers District Engineer for the Charleston District implements the Clean Water Act Section 404 and the Rivers and Harbors Act permitting program for SRS spent nuclear fuel construction activities that would affect U.S. waters.

In accordance with the Endangered Species Act, the SRS would consult with the U.S. Fish and Wildlife Service, Charleston Field Office on impacts that spent nuclear fuel construction activities could have on threatened and endangered species.

2.2.2 State

The South Carolina Department of Health and Environmental Control implements the following State laws that would affect SRS spent nuclear fuel management activities:

- Pollution Control Act (nonradioactive emissions and discharges, and nonhazardous waste management)
- Hazardous Waste Management Act (nonradioactive hazardous waste management)
- Safe Drinking Water Act
- Groundwater Use Act
- Stormwater Management and Sediment Reduction Act

The U.S. Army Corps of Engineers District Engineer for the Charleston District has an agreement with the South Carolina Department of Health and Environmental Control whereby that department issues Clean Water Act Section 401 water quality certifications. The South Carolina Department of Health and Environmental Control also receives SRS reports in accordance with the Emergency Planning and Community Right-To-Know Act.

The South Carolina State Archives Department includes the State Historic Preservation Officer. In accordance with the National Historic Preservation Act, the SRS would consult with this department on impacts that construction activities could have on cultural resources.

2.2.3 Local

The only local requirement applicable to SRS spent nuclear fuel management is the Aiken County Sediment Control Ordinance, which would affect construction activities.

2.3 Spent Nuclear Fuel Management Program at the Savannah River Site

This EIS addresses the management of approximately 2,759 metric tons of heavy metal (MTHM; 3,058 tons) of spent nuclear fuel stored at various locations within the DOE Complex. At present, DOE has stored approximately 201.5 MTHM (222.1 tons), or about 7 percent of this material, at the SRS. The spent nuclear fuel currently stored at the SRS that DOE has included in the analyses in this document includes:

- 184.4 MTHM (203.3 tons) of Savannah River Defense Production [highly enriched uranium (HEU) aluminum-clad fuels], including plutonium target material
- 4.6 MTHM (5.1 tons) of commercial spent fuel (primarily zirconium-clad)
- 11.9 MTHM (13.1 tons) of test and experimental reactor Zircaloy-clad fuel
- 0.6 MTHM (0.7 ton) of test and experimental reactor stainless-steel-clad fuel

The F- and H-Area Canyons at the SRS are among the only remaining operable chemical separations facilities of their kind in the DOE Complex. Each canyon has an associated storage basin that serves as an interim staging area where reactor fuel bundles and targets await the Chemical Separations Process. The basins currently contain 13 reactor fuel assemblies (H-Area) and aluminum-clad targets (F-Area).

DOE has stored most of the remaining aluminum-clad spent nuclear fuel from SRS reactor operations under water in concrete reactor storage basins. Three reactor disassembly basins (K-, P-, and L-Reactors) contain reactor fuel and target material. These structures were built in the 1950s and were not intended for the prolonged storage of radioactive materials. Wet (underwater) storage, while potentially viable for stainless-steel-clad fuel elements, is not satisfactory for aluminum-clad elements, which are subject to corrosion and pitting.

In March 1992, chemical processing operations were suspended in the canyons to address a potential safety concern. The concern was subsequently addressed but prior to resumption of processing, the Secretary of Energy directed that defense related chemical separations activities (i.e., reprocessing) be phased out at the SRS. Since the decision, DOE has determined that further action related to the disposition of nuclear material, including spent nuclear fuel, is subject to the National Environmental Policy Act (NEPA) process. Non-safety related facility operations have remained shut down with the exception of Pu-238 processing associated with the support of NASA missions.

As a result of these shut-downs, the canyons and the basins used for storage of spent nuclear fuel and irradiated targets have a large inventory of in-process solutions and fuel and targets (respectively). Some materials stored in the L- and K-Reactor disassembly basins have corroded, releasing fissile materials to the pool water. DOE is preparing an environmental impact statement that will evaluate risks that these and other SRS materials represent to the public and workers and will assess the near-term need for actions to stabilize these materials to ensure continued safe management (*Notice of Intent to Prepare an Environmental Impact Statement for the Interim Management of Nuclear Materials at the Savannah River Site*, 59 FR 12588, 3/17/94). These actions would take place over the short-term (about 5 years), until DOE can make programmatic decisions on disposition.

DOE stores other spent fuel in the Receiving Basin for Offsite Fuels (RBOF) on the SRS. This basin, which is in H-Area near the center of the Site, has been operating and receiving fuels of U.S. origin since 1964. This 15,000-square-foot (1,393-square-meter) facility consists of an unloading basin, two storage basins, a repackaging basin, a disassembly basin, and an inspection basin. The basins and their interconnecting transfer canals hold about 500,000 gallons (1,893,000 liters) of water. Spent fuel elements arrive in lead-lined casks weighing from 24 to 70 tons (about 22 to 64 metric tons), which a crane lifts from a railroad car or truck trailer and places in the unloading basin. About 30 percent of the fuels in the Receiving Basin for Offsite Fuels consist of uranium clad in stainless steel or Zircaloy, which SRS facilities cannot process without modifications.

2.4 Vulnerabilities Associated with SRS Spent Nuclear Fuel

In August 1993, the Secretary of Energy commissioned a comprehensive baseline assessment of the environmental, safety, and health vulnerabilities associated with the storage of spent nuclear fuel in the DOE complex. The purpose of this assessment was to determine the inventory and condition of the Department's Reactor Irradiated Nuclear Material, which includes spent nuclear fuel and reactor irradiated target material. The assessment also evaluated the condition of the facilities that store spent fuel and identified the vulnerabilities and problems currently associated with these facilities. Vulnerabilities in nuclear facilities are conditions or weaknesses that could lead to radiation exposure to the public, unnecessary or increased exposure to workers, or release of radioactive materials to the environment. Loss of institutional controls, such as a cessation of facility funding or reductions in facility maintenance and control, could cause some vulnerabilities.

Based on this evaluation process DOE released a report to the Secretary of Energy, entitled *Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and other Reactor Irradiated Nuclear Materials and Their Environmental, Safety and Health Vulnerabilities* (i.e., "The Working Group Report," Volumes I, II, and III), to the public on December 7, 1993 (DOE 1993). This report identified 106 vulnerabilities associated with spent fuel storage in the DOE complex, including 21 at the Savannah River Site. The report also determined that five facilities and three burial grounds warranted priority attention from management to avoid unnecessary increases in worker radiation exposure and cost during cleanup. The Savannah River Site L- and K-Reactor Disassembly Basins were among these facilities. The report grouped vulnerabilities associated with each facility into three categories for management attention based on when corrective action should be initiated: less than 1 year, 1 to 5 years, and more than 5 years.

After issuing the Working Group Report, DOE developed a Plan of Action to address all vulnerabilities, taking into consideration currently available resources for implementation. The Plan of Action is a consolidation of individual action plans designed to address each spent nuclear fuel vulnerability in a manner that reflects the DOE (1) sense of urgency, (2) concern for worker protection, (3) commitment to mitigate environmental impacts, and (4) need for compatible long-term solutions.

The interim goal for the Savannah River Site reactor disassembly basins, pending completion of the removal of the stored material, is the stabilization of basin conditions to reduce corrosion and to address known vulnerabilities. The long-term goal of the action plan is a safe start of the removal of reactor-irradiated nuclear material within a 5-year period, consistent with safe and environmentally sound operations, including completion of appropriate National Environmental Policy Act (NEPA) review. These actions will lead to mitigating the identified vulnerabilities while DOE pursues other courses of action.

The 21 vulnerabilities identified for the Savannah River Site now have complete Actions Plans (DOE, 1994a, 1994b). Table 2-2 lists SRS vulnerabilities by facility, tracking number, priority categorization, and Action Plan status.

DOE is currently implementing a number of the 21 Action Plans. These actions have been evaluated under the NEPA review process. The remaining corrective actions, those that will be carried out through FY99, will also undergo NEPA review prior to implementation. Of these outstanding actions, only the construction of a dry storage facility would require detailed NEPA documentation (e.g., an EIS). The construction of such a facility is programmatically addressed in this EIS, but would require a site-specific NEPA evaluation prior to implementation.

2.5 Representative Host Sites

DOE has identified two SRS areas as representative host sites for potential facilities related to the implementation of programmatic decisions on spent nuclear fuel management (Figure 2-3):

- F- and H-Areas (considered together) for the modification or expansion of existing facilities, new wet storage, and support facilities

- An undeveloped site for the construction of major new facilities, primarily an Expended Core Facility or dry storage vault

2.5.1 F- and H-Areas

These two areas contain most of the current spent nuclear fuel facilities and operations at the SRS, including the Receiving Basin for Offsite Fuels. Therefore, DOE would focus future actions

Table 2-2. SRS vulnerabilities by facility, vulnerability, tracking number, priority categoriz. on, and Action Plan status.

Site/Facility Vulnerability Number Description	Priority: Eight major facilities with vulnerabilities	Priority: Less than 1 year	Priority: Greater than 1 year	Action Plan status
SRS/L-Reactor Disassembly Basin SRS-01 Potential unmonitored buildup of radionuclide or fissile materials in sand filters.	✓			Complete
SRS/L-Reactor Disassembly Basin SRS-03 Different load bearing bolts installed in I-beam RINM and target hanger trolleys.		✓		Complete
SRS/L-Reactor Disassembly Basin SRS-04 Lack of authorization basis in operating the sand filter cleanup system for L-Area Disassembly Basin.	✓			Complete
SRS/Reactor Disassembly Basins SRS-05 Corrosion of aluminum clad fuel, targets, and components.			✓	Complete
SRS/L-Reactor Disassembly Basins SRS-06 Cesium-137 activity level in L-Basin.	✓			Complete
SRS/L-Reactor Disassembly Basins SRS-07 Determine whether gas bubbles release is a potential hazard above the bucket storage area at L-Reactor.	✓			Complete
SRS/K-, L-, P-Reactors SRS-08 Lack of Reactor Authorization Basis.	✓			Complete
SRS/K-, L-Reactor Disassembly Basins SRS-09 Corrosion of Mark 31 A and B target slugs in K and L disassembly basins.	✓			Complete
SRS/P-Reactor Disassembly Basins SRS-10 Hoist Rod Corrosion.		✓		Complete
SRS/K-, L-Reactor Disassembly Basins SRS-11 Reactor Disassembly Basin Safety Analysis Envelope.	✓			Complete
SRS/L-Reactor Disassembly Basin SRS-12 Inadvertent flooding of L-Reactor Disassembly Basin.	✓			Complete

Table 2-2. (continued).

Site/Facility Vulnerability Number Description	Priority: Eight major facilities with vulnerabilities	Priority: Less than 1 year	Priority: Greater than 1 year	Action Plan status
SRS/K-Reactor Disassembly Basin SRS-13 Inadvertent flooding of K-Reactor Disassembly Basin.	✓			Complete
SRS/P-Reactor Disassembly Basin SRS-14 Inadvertent flooding of P-Reactor Disassembly Basin.	✓			Complete
SRS/RBOF; P-, R-, L-, C-, R-Reactors SRS-15 (NOTE: RBOF is a less than 1 year vulnerability) Conduct of operations at reactor facilities and RBOF.	✓			Complete
SRS/Receiving Basin for Offsite Fuel (RBOF) SRS-16 Inadequate tornado protection at RBOF.		✓		Complete
SRS/Receiving Basin for Offsite Fuel (RBOF) SRS-17 Seismic vulnerability of RBOF.		✓		Complete
SRS/H-Area Canyon SRS-18 Seismic vulnerability of H-Area Canyon.			✓	Complete
SRS/F-Area Canyon SRS-19 Seismic vulnerability of F-Area Canyon.			✓	Complete
SRS/K-, L-, P-Reactor Disassembly Basins and RBOF SRS-20 Inadequate leak detection system in the underground water-filled RINM storage basin.		✓		Complete
SRS/L-, K-, P-Reactor Disassembly Basins SRS-21 Inadequate seismic evaluation and potential inadequacies of structures, systems, and components to withstand a design basis event.	✓			Complete
SRS/Area R SRS-22 Potential buried Spent Nuclear Fuel.		✓		Complete

under any of the alternatives in these areas as well, for cost-effectiveness and because construction would occur in areas that had been previously disturbed.

F- and H-Areas are about 2 miles (3.2 kilometers) apart near the center of the SRS. The nearest Site boundary is approximately 7.5 miles (12 kilometers) to the west. DOE uses the land within a 5-mile (8-kilometer) radius of the two areas either for industrial purposes associated with SRS operations or as managed forest land. The closest facility to F- and H-Areas is the E-Area Solid

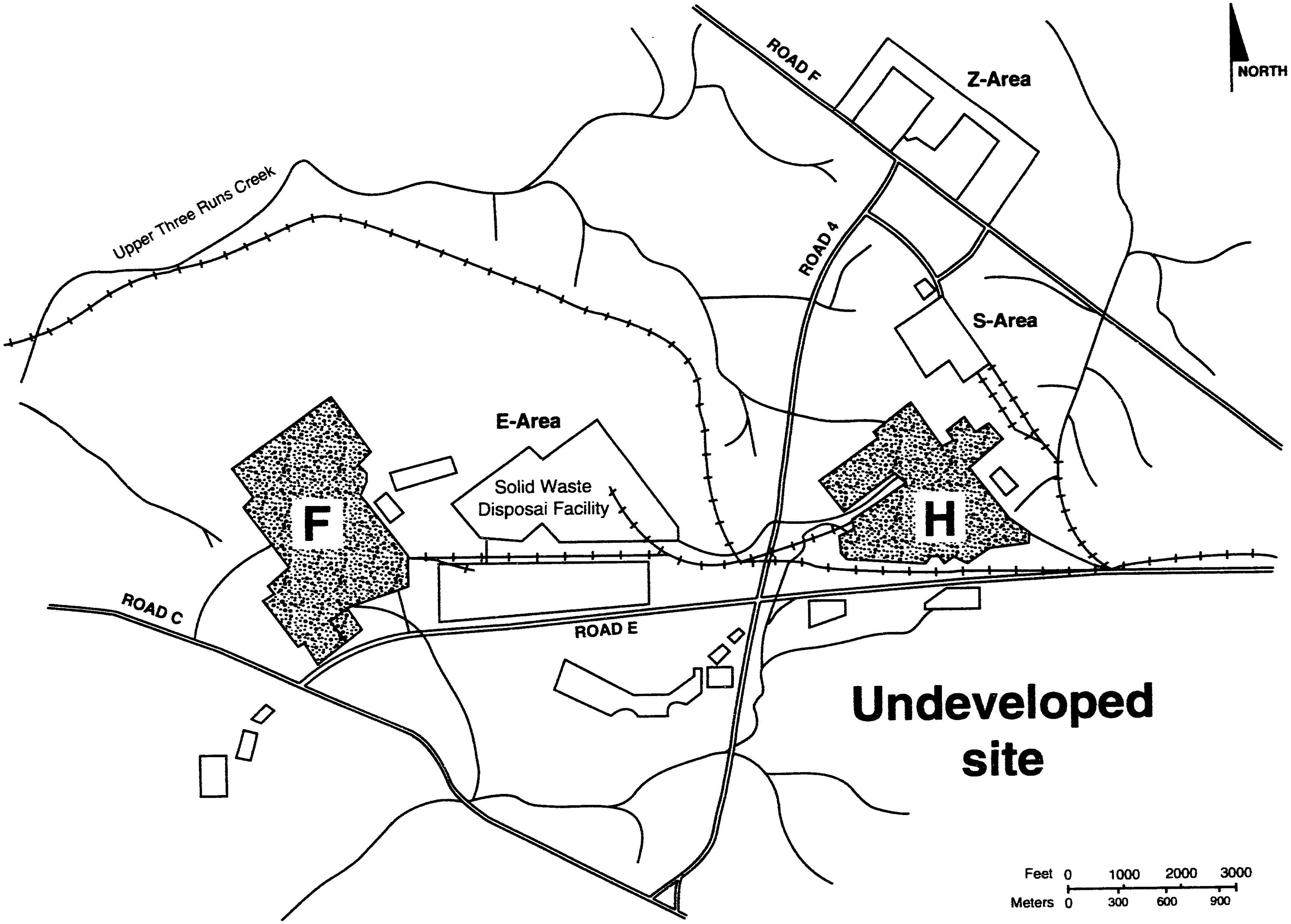

Figure 2-3. Representative host sites on Savannah River Site.

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Waste Disposal Facility, which lies between the two areas (Figure 2-3). DOE uses this facility to dispose of SRS solid low-level radioactive waste and to store TRU radioactive waste and mixed waste.

The F-Area separations facilities occupy about 420 acres (1.7 square kilometers). These facilities were designed primarily for the recovery of plutonium-239 from irradiated and unirradiated feed materials. DOE used the F-Area Canyon to dissolve target materials and produce solutions that contained the various products extracted from fission products. Further processing converted the products from solution to solid form for shipment off the Site. Large tanks in F-Area store high-level liquid radioactive waste for future stabilization and disposal through the Defense Waste Processing Facility.

H-Area facilities occupy about 395 acres (1.6 square kilometers). The H-Area Canyon processed irradiated fuel elements or target assemblies from reactors. Primary operations included the dissolution of irradiated targets and fuel tubes, chemical and physical separation, and purification of materials. DOE stores high-level liquid waste in large tanks in H-Area, as in F-Area, for future processing and disposal through the Defense Waste Processing Facility.

2.5.2 Undeveloped Representative Host Site

DOE has selected an undeveloped representative host site for the construction of new facilities that F- or H-Area could not accommodate. This site is to the south and east of H-Area, adjacent to SRS Road E and close to an existing railroad line, as shown in Figure 2-3. The SRS could make connections to existing electricity, water, and steam networks with minimal additional construction. The use of this site would have the advantage of consolidating spent nuclear fuel-related activities near F- and H-Areas and close to the center of the SRS.

This site is representative of many available areas on the SRS that could support spent nuclear fuel management activities. For example, DOE has identified a different representative site for the possible construction of the Expended Core Facility for the management of naval spent nuclear fuel (see Appendix D of this Environmental Impact Statement). DOE would conduct a detailed siting analysis before implementing any programmatic decision at the SRS. DOE would assess, as necessary, the environmental consequences of the siting of any facilities as part of the site-specific NEPA documentation.

3. SPENT NUCLEAR FUEL ALTERNATIVES

This chapter describes the five management alternatives for spent nuclear fuel that the U.S. Department of Energy (DOE) has evaluated for the Savannah River Site (SRS) as part of Volume 1 of this Environmental Impact Statement. These alternatives are:

1. No Action
2. Decentralization
3. 1992/1993 Planning Basis
4. Regionalization (with 3 subalternatives for the SRS)
5. Centralization (with 2 subalternatives for the SRS)

The activities covered by the alternatives range from maintaining the current inventory of spent fuel at the SRS without receiving any more shipments (Alternative 1), through keeping the existing inventory and accepting or sending off some limited shipments (Alternatives 2 through 4), to receiving at the Site all DOE spent nuclear fuel and some from other sources (Alternative 5). DOE also examined an option for shipping all spent nuclear fuel at the SRS to another location (a variation of Alternatives 4 and 5). Table 3-1 summarizes the quantities of material that would be received, shipped out, and ultimately managed at the SRS under the various alternatives. DOE has assessed the aluminum-clad spent nuclear fuel separately from nonaluminum-clad fuel (i.e., stainless-steel and Zircaloy) because the options for managing them at the Site could be different as explained in Section 3.1.

The analytical approach used in this document produces estimates of consequences that would be as large as or larger than any that could occur or be expected under the alternatives and provides a comparison of the impacts of the principal technologies for managing spent nuclear fuel at the SRS.

This chapter also provides an overview of the SRS management approach and describes the five alternatives as they relate to the SRS (Sections 3.1 and 3.2). In addition, the chapter summarizes and compares the potential environmental consequences of each alternative (Section 3.3).

Table 3-1. Quantities (MTHM)[a] of spent nuclear fuel that would be received, shipped, and managed at the SRS under the five alternatives.[b,c]

Alternative	Fuel Type	Currently at SRS	Receive	Ship Out	Totals managed at SRS under this alternative
1. No Action	Aluminum	184.40	0.00	0.00	184.40
	Nonaluminum	17.10	0.00	0.00	17.10
	Totals	201.50	0.00	0.00	201.50
2. Decentralization	Aluminum	184.40	8.22	0.00	192.62
	Nonaluminum	17.10	1.13	0.00	18.23
	Totals	201.50	9.35	0.00	210.85
3. 1992/1993 Planning Basis	Aluminum	184.40	11.13	0.00	195.53
	Nonaluminum	17.10	3.52	0.00	20.62
	Totals	201.50	14.65	0.00	216.15
4. Regionalization - A (by fuel type)	Aluminum	184.40	23.29	0.00	207.69
	Nonaluminum	17.10	0.00	(17.10)	0.00
	Totals	201.50	23.29	(17.10)	207.69
4. Regionalization - B (by location at SRS)	Aluminum	184.40	15.20	0.00	199.60
	Nonaluminum	17.10	29.87	0.00	46.97
	Totals	201.50	45.07	0.00	246.57
4. Regionalization - B (by location, elsewhere)	Aluminum	184.40	0.00	(184.40)	0.00
	Nonaluminum	17.10	0.00	(17.10)	0.00
	Totals	201.40	0.00	(201.50)	0.00
5. Centralization (at SRS)	Aluminum	184.40	23.29	0.00	207.69
	Nonaluminum	17.10	2,533.81	0.00	2,550.91
	Totals	201.50	2,557.10	0.00	2,758.60
5. Centralization (elsewhere)	Aluminum	184.40	0.00	(184.40)	0.00
	Nonaluminum	17.10	0.00	(17.10)	0.00
	Totals	201.50	0.00	(201.50)	0.00

a. To convert metric tons of heavy metal to tons, multiply by 1.1023.
b. Numbers may not sum due to rounding.
c. Source: Wichmann (1994).

3.1 SRS Management Approach

3.1.1 Management Options

DOE has evaluated three options for the management of spent nuclear fuel at the SRS under the five alternatives considered for this EIS. These options are wet storage or dry storage of all fuels and the processing of aluminum-clad fuels. DOE could implement these options individually or in combination under any of the five alternatives. However, the level of analysis in this EIS is insufficient to allow selection of a particular option. DOE would base its selection of one or more management options on additional analysis, including a separate SRS-specific NEPA review based on this programmatic EIS.

3.1.1.1 Wet Storage. As described above in Section 2.3, the SRS currently maintains its spent nuclear fuel in wet storage in the Receiving Basin for Offsite Fuels and several reactor basins. Wet storage under the 40-year interim management plan (except under the No-Action alternative) would require that DOE construct a new wet storage pool at the SRS and move all fuel to this facility. Prior to this transfer, DOE could place all the aluminum-clad fuel in stainless steel canisters to prevent further corrosion and breakdown of the fuel cladding. The stainless-steel- and Zircaloy-clad fuels could also require canning. The SRS would monitor and maintain the water quality and the condition of the fuel in the storage pool throughout the interim management period.

Under this wet storage option, the spent nuclear fuel would be in an interim storage form, which could require further treatment depending on the DOE decision on its ultimate disposition.

3.1.1.2 Dry Storage. DOE currently has no dry storage facilities for spent nuclear fuel at the Site. Dry storage of SRS aluminum-clad fuels under this management plan would require technology development prior to the construction of a dry storage facility. Although such facilities exist at other DOE sites and at commercial locations, DOE believes that the characteristics of SRS spent fuel are sufficiently different to require some research and development before the design and construction of a facility for this fuel. DOE would can all fuel before placing it into the dry storage vaults. It would also have to maintain and monitor the facility for the remainder of the 40-year management period.

As with wet storage, the dry storage option would place the spent fuel into an interim storage form that could require further treatment later depending upon DOE's decision on ultimate disposition.

3.1.1.3 Processing and Dry Storage. One method under this option would be for the SRS to process existing aluminum-clad spent nuclear fuel through the existing separations facilities in the F- and H-Area Canyons, and place the nonaluminum-clad fuels and any future receipts in dry storage. The process using existing capability would result in the generation of both separated actinides (e.g., uranium oxide), which would be stored on the site in existing facilities, and solutions of fission products that would be placed in existing waste storage facilities for later conversion to a glassified form through the Defense Waste Processing Facility (DWPF). DOE would maintain and monitor the dry storage facility containing the non-aluminum spent fuel. Variations of this processing option are also possible, such as processing all the aluminum-clad fuel currently on the Site plus all that is received from elsewhere, or developing the capability at the SRS for processing for vitrification without chemical separations.

The process option selected for evaluation in this document is representative of possible processing options that might be employed, but is not necessarily the one that DOE would select. Detailed National Environmental Policy Act evaluations would be required to implement any spent nuclear fuel management plan at the SRS.

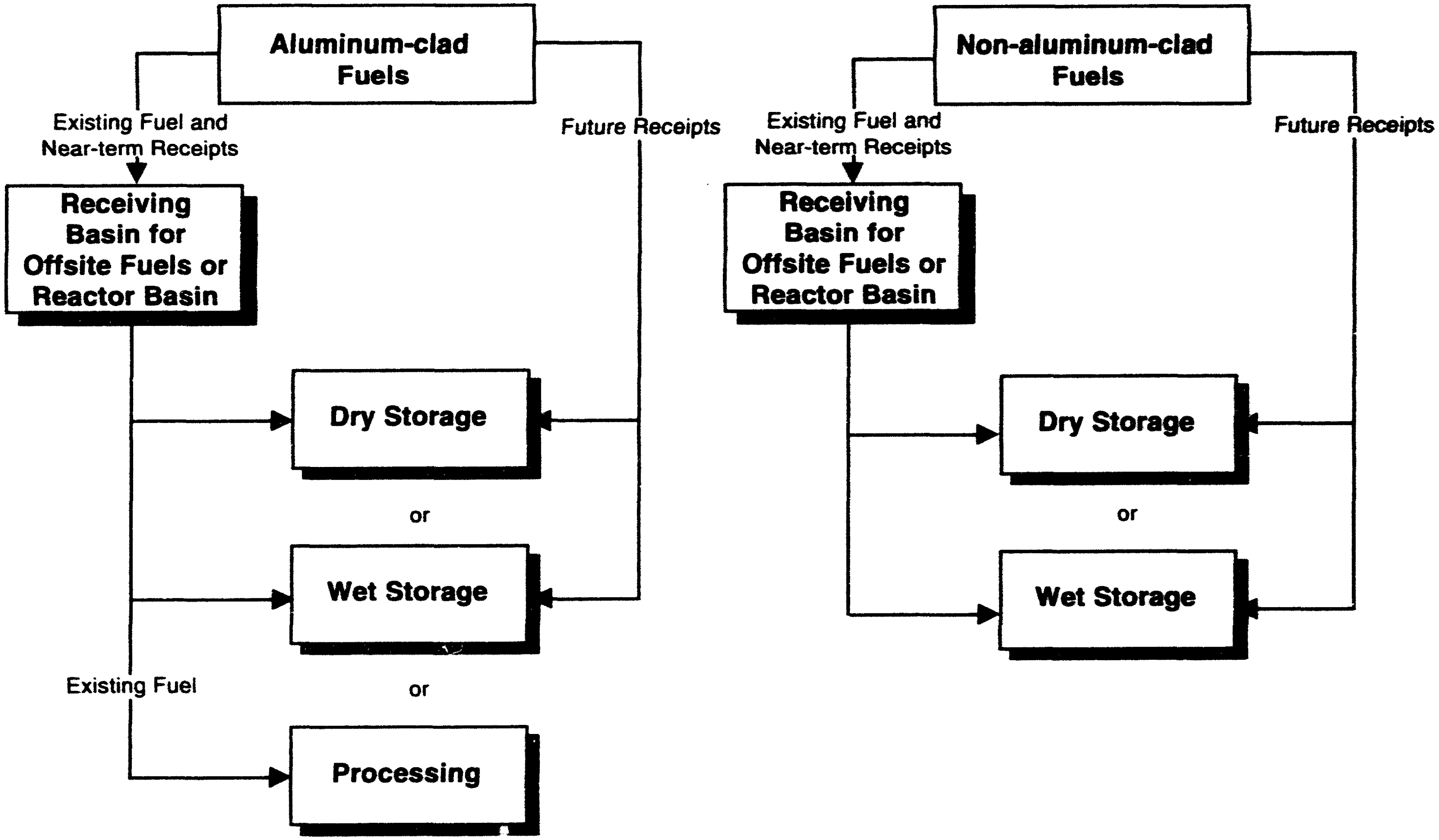
3.1.2 Management Plan

Figure 3-1 summarizes DOE's overall plan for the interim management of aluminum-clad and nonaluminum-clad fuels at the SRS. This flowchart shows actions for all alternatives except No Action, as explained in Section 3.2.1.

3.1.2.1 Aluminum-clad Fuels. Depending on the alternative and option selected, DOE could (within constraints of mission commitments) consolidate some aluminum-clad fuel in the Receiving Basin for Offsite Fuels to take advantage of this facility's superior water quality and then move all aluminum-clad fuel into dry storage, wet storage, or initiate processing (Figure 3-1). DOE could also process aluminum-clad fuel without any consolidation work. Before moving the fuel into dry or wet storage, DOE would place it in cans. DOE would hold the canned fuel or the stabilized products from processing in storage for the 40-year interim management period until it decided their final disposition.

DOE would place aluminum-clad fuels received by the SRS from other locations in wet or dry storage. DOE could not implement any of the options for aluminum-clad fuels, with the exception of processing using existing SRS capabilities, without a technology development effort.

3.1.2.2 Nonaluminum-clad Fuels. DOE options for the management of nonaluminum-clad fuels at the SRS are somewhat different, in that only dry or wet storage is considered (Figure 3-1). The processing of these fuels at the Site is not an option because the SRS does not currently have operational facilities capable of separating these materials. To improve aluminum-clad fuel storage, DOE could consolidate the nonaluminum-clad fuel inventory in a reactor basin where the more resistant stainless-steel or Zircaloy cladding would be less susceptible to corrosion. The fuel would remain there until DOE built new dry or wet storage facilities. DOE would then can the fuel and move it into the new storage. DOE would place any nonaluminum-clad fuel received at the SRS after completion of the new facilities directly into storage. The fuel would remain in this interim storage until DOE decided its ultimate disposition.



SFIG 0301

Figure 3-1. Diagram of how SRS would manage aluminum-clad and nonaluminum-clad fuels. "Near-term Receipts" refers to the fuel that would be received before new wet or dry storage facilities are available.

3.2 Description of Alternatives

3.2.1 Overview

Table 3-2 compares actions under each of the five alternatives. These actions relate to the requirements for transportation, stabilization, facilities, and research and development that DOE would address for each alternative. Transportation would include onsite movements as well as the receipt or shipment of spent fuel. The consideration of facilities addresses not only new ones that could be required, but also the use of existing structures and capabilities such as the F- and H-Area Canyons at SRS. Finally, each alternative would involve some level of research and development on matters related to spent nuclear fuel interim management (e.g., stabilization, transportation casks) and its ultimate disposition.

Alternative 1 (No Action) addresses only the interim wet storage option, while the analysis of Alternatives 2 through 5 considers three options: dry storage, wet storage, and processing of existing aluminum-clad fuels and placing the other fuels into storage. In addition, Alternatives 4 and 5 include an option for the shipment of spent nuclear fuel off the SRS. This analytical approach shows the relative impact of viable interim storage technologies for the range of alternatives this EIS is considering for the SRS. However, this information is not sufficient to support the selection of a specific interim storage technology at the SRS because DOE has not completed site-specific research and development for dry storage and wet storage methods or an evaluation of other processing options. In addition, the specific quantities of offsite fuel that DOE would manage are subject to change. The selection of an interim storage technology will be the subject of separate National Environmental Policy Act documentation specific to the SRS.

Figure 3-2 is a matrix showing the types of facilities that would be required for each alternative and option. The list includes those facilities already operating at the SRS (e.g., Receiving Basin for Offsite Fuels) as well as potential facilities (e.g., fuel characterization facility). DOE considered these facilities in its evaluation of the consequences of each alternative, as described in Chapter 5.

The alternatives described below address interim storage to 2035; further treatment of the spent nuclear fuel would be necessary before DOE obtained a final disposable waste form. This EIS does not address this additional treatment. However, DOE would carry out a full National Environmental Policy Act documentation for any decision on final disposition of spent nuclear fuel.

Table 3-2. Actions required under each of the five alternatives at the SRS.

Alternative	Transportation	Stabilization	Facilities	Research and Development
1. No Action	No shipments to or from the Site. Limit onsite transfers to those required for safe storage.	Place aluminum-clad fuels that are badly corroded and in danger of cladding failure in containers and return them to wet storage.	Store fuels in Receiving Basin for Offsite Fuels and in an upgraded reactor basin. Requires no new facilities.	Continue existing spent nuclear fuel-related research and development.
2. Decentralization	Receive about 9.4 MTHM (10.4 tons) of aluminum-clad and nonaluminum-clad fuels. Limit onsite transfers to those required for safe storage, consolidation, and research and development. Later relocate fuels to new wet or dry storage facility or move aluminum-clad fuels to F- and H-Canyons for processing.	Can aluminum-clad fuels and place them in wet or dry storage or process existing fuel through F- and H-Canyons. Can stainless-steel and Zircaloy fuels and place in wet or dry storage.	Store fuels in Receiving Basin for Offsite Fuels or upgraded reactor basin until new wet or dry storage facility is built. Requires new characterization facility, new wet or dry canning facility, and new wet or dry storage facility.	Develop technology (canning and storage design) to store SRS aluminum-clad fuels in dry storage vault. Conduct research and pilot-scale operations to determine best technology for ultimate disposition of aluminum-clad fuels.
3. 1992/1993 Planning Basis	Receive about 14.6 MTHM (16.1 tons) of aluminum-clad and nonaluminum-clad fuels. Limit onsite transfers to those required for safe storage, consolidation, and research and development. Later relocate fuels to new wet or dry storage facility, or move aluminum-clad fuels to F- and H-Canyon for processing.	Can aluminum-clad fuels and place them in wet or dry storage or process existing fuel through F- and H-Canyons. Can stainless steel and Zircaloy fuels and place in wet or dry storage.	Store fuels in Receiving Basin for Offsite Fuels or upgraded reactor basin until new wet or dry storage facility is built. Requires new characterization facility, new wet or dry canning facility and new wet or dry storage facility.	Develop technology (canning and storage design) to store SRS aluminum-clad fuels in dry storage vault. Conduct research and pilot-scale operations to determine best technology for ultimate disposition of aluminum-clad fuels.
4. Regionalization - A (by fuel type at the SRS)	Receive about 23.3 MTHM (25.7 tons) of aluminum-clad fuel. Ship to Idaho National Engineering Laboratory about 17.1 MTHM (18.8 tons) of stainless steel and Zircaloy fuel. Relocate aluminum-clad fuels to Receiving Basin for Offsite Fuels, as necessary; then to new wet or dry storage facilities, or move aluminum-clad fuels to F- and H-Canyon for processing.	Can aluminum-clad fuels and place them in wet or dry storage; or process existing fuel through F- and H-Canyons.	Store fuel in existing Receiving Basin for Offsite Fuels or upgraded reactor basin until new wet or dry storage facility is available, or until fuel is processed. Requires new receiving and characterization facilities, new wet or dry canning facilities, and new wet or dry storage facilities.	Develop technology (canning and storage design) to store aluminum-clad fuels in dry storage vault. Conduct research and pilot-scale operations to determine best technology for ultimate disposition of aluminum-clad fuels.

Table 3-2. (continued).

Alternative	Transportation	Stabilization	Facilities	Research and Development
4. Regionalization - B (by location at the SRS)	Receive approximately 45.1 MTHM (49.7 tons) of spent fuel from other locations. Limit onsite transfers to those required for safe storage, consolidation, and research and development. Relocate fuels to new dry or wet storage facility or move aluminum-clad fuel to F- and H-Canyons for processing.	Can aluminum-clad fuels and place them in wet or dry storage; or process existing aluminum-clad fuels through F- and H-Canyons and store remaining fuel. Characterize and can fuel received from offsite that is not in a form suitable for direct placement into storage.	Store fuels in Receiving Basin for Offsite Fuels or upgraded reactor basin until new storage facility is available. Store new fuel shipments in new wet or dry storage facility. Requires new receiving, characterization and canning facilities, new wet or dry storage facility, and possibly a new Expended Core Facility.	Develop technology (canning and storage design) to store SRS aluminum-clad fuels in dry storage vault. Conduct research and pilot-scale operations to determine best technology for ultimate disposition of aluminum-clad fuels.
4. Regionalization - B (by location at another site)	Move all fuels to new characterization facility prior to shipment offsite. Ship out about 201.4 MTHM (222.0 tons) of spent fuel.	Characterize and can all spent fuel prior to shipment.	Store existing fuels in Receiving Basin for Offsite fuel and in a reactor basin until characterization and shipment offsite. Requires new characterization facility.	Develop technology for stabilization, canning, and shipment of degraded aluminum-clad fuel.
5. Centralization (at the SRS)	Receive about 2,557.1 MTHM (2,818.7 tons) of spent fuel from offsite. Limit onsite transfers to those required for safe storage, consolidation, and research and development. Relocate fuels to new dry or wet storage facility or move aluminum-clad fuel to F- and H-Canyons for processing.	Can aluminum-clad fuels and place them in wet or dry storage; or process existing aluminum-clad fuels through F- and H-Canyons and store remaining fuels. Characterize and can fuel received from offsite that is not in a form suitable for direct placement in storage.	Store fuel in Receiving Basin for Offsite Fuels or in an upgraded reactor basin until new storage facilities are available. Store new fuel shipments in new wet or dry storage facility. Requires new receiving, characterization and canning facilities, new wet or dry storage facility, and new Expended Core Facility.	Develop technology (canning and storage design) to store SRS aluminum-clad fuels in dry storage vault. Conduct research and pilot-scale operations to determine best technology for ultimate disposition of spent nuclear fuels.
5. Centralization (at another site)	Move all fuels to new characterization facility prior to shipment offsite. Ship out about 201.4 MTHM (222.0 tons) of spent fuel.	Characterize and can all spent fuel prior to shipment.	Store existing fuel in Receiving Basin for Offsite Fuel or in an upgraded reactor basin until characterization and shipment offsite. Requires new characterization facility.	Develop technology for stabilization, canning, and shipment of degraded aluminum-clad fuel.

Facility	No Action	Decentralization			1992/93 Planning Basis			Regionalization - A (by Fuel Type)		
	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c
	Wet	Dry	Wet	Process[c]	Dry	Wet	Process[c]	Dry	Wet	Process[c]
Reactor Basins	●	●	●	●	●	●	●	●	●	●
Receiving Basin Offsite Fuels	●	●	●	●	●	●	●	●	●	●
New Fuel Characterization		○	○	○	○	○	○	○	○	
New Dry Canning		○		○	○		○	○		
New Interim Dry Storage		○		○	○		○	○		
New Expended Core (Navy)										
New Fuel Receiving		○	○	○	○	○	○	○	○	○
New Wet Canning[b]			○			○			○	○
New Fuel Storage Pool			○			○			○	○
H-Canyon/H-Area Separations	X			●			●			●
F-Canyon/F-Area Separations	X			●			●			●

Facility	Regionalization - B (by Location)				Centralization			
	Option 4d	Option 4e	Option 4f	Option 4g	Option 5a	Option 5b	Option 5c	Option 5d
	Dry	Wet	Process[c]	Ship	Dry	Wet	Process[c]	Ship
Reactor Basins	●	●	●	●	●	●	●	●
Receiving Basin Offsite Fuels	●	●	●	●	●	●	●	●
New Fuel Characterization	○	○	○	○	○	○	○	○
New Dry Canning	○		○		○		○	
New Interim Dry Storage	○		○		○		○	
New Expended Core Facility (Navy)	✱	✱	✱		○	○	○	
New Fuel Receiving	○	○	○		○	○	○	
New Wet Canning[b]		○				○		
New Fuel Storage Pool		○				○		
H-Canyon/H-Area Separations			●				●	
F-Canyon/F-Area Separations			●				●	

Legend:

○ New facilities required under each case

● Existing facilities required under each case

X Existing facilities that would be involved to maintain safe storage

✱ May be needed

a. Information derived from WSRC (1994).

b. Includes fuel repackaging facility.

c. Option includes processing of existing aluminum-clad fuels and storage of others.

Figure 3-2. Types of facilities required for each alternative.[a]

SFIG 0302

3.2.2 Alternative 1 - No Action

3.2.2.1 Overview. This alternative deals only with the minimum actions that DOE would deem necessary for the continued safe and secure management of spent nuclear fuel. It is not a *status quo* condition. Rather, across its complex of facilities, DOE would maintain spent nuclear fuel close to generation or current storage locations with no shipment between sites. Facility upgrades or replacements and onsite fuel transfers would occur only to support safe and secure interim storage. DOE would continue existing and new research and development activities for spent fuel interim management. Stabilization activities would be limited only to those minimum actions required to store spent nuclear fuel safely.

3.2.2.2 SRS Alternative 1 - Wet Storage. DOE would initiate the various SRS programs and activities necessary to obtain optimum use of existing spent nuclear fuel facilities for the extended storage of existing Site inventories totalling 201.5 metric tons (222.1 tons) of heavy metal (MTHM) in the following quantities:

- 184.4 MTHM (203.3 tons) of Savannah River Defense Production [highly enriched uranium (HEU) aluminum-clad fuels], including plutonium target material

- 5.2 MTHM (5.7 tons) of test and experimental reactor stainless-steel-clad fuel

- 11.9 MTHM (13.1 tons) of test and experimental reactor Zircaloy-clad fuel

The goal of this program would be to relocate some aluminum-clad fuels to the Receiving Basin for Offsite Fuels where precisely maintained water quality would prolong the storage life of these fuel types. In addition, DOE would relocate a portion of the stainless-steel and Zircaloy fuels to a reactor basin, where their more resistant cladding would maintain fuel containment for an extended period. These actions would be accomplished within the constraints of mission requirements.

The following describes one method that could be employed to improve the storage of aluminum-clad fuel. Variations of this plan that would involve only the use of existing storage basins are also possible.

- Select a reactor basin for upgrading and for the interim storage of SNF.

- Relocate aluminum-clad fuels from the selected reactor basin to other onsite basins to enable cleaning and repair of the basin chosen for upgrade to improve water quality.

- Consolidate fuels in the Receiving Basin for Offsite Fuels to the extent possible.

- After cleaning and renovating the selected reactor basin, move a portion of the stainless-steel and Zircaloy-clad fuel assemblies now at the Receiving Basin for Offsite Fuels to the renovated reactor basin.

- Move the aluminum-clad fuels temporarily stored at other locations to the Receiving Basin for Offsite Fuels or the renovated reactor basin.

DOE will continue to place heavily corroded aluminum-clad fuel elements that could be in danger of cladding failure into containers in the wet pool as required to minimize any spread of materials throughout the pool. This action would be much simpler than canning the elements, which would occur under the other alternatives.

This alternative would require no new facilities. DOE would continue existing spent nuclear fuel-related research and development.

3.2.3 Alternative 2 - Decentralization

3.2.3.1 Overview. Under this alternative, DOE would maintain existing spent nuclear fuel in storage at the current locations, and the SRS would receive some shipments of university fuel and foreign fuel. This alternative differs from the No-Action alternative by allowing significant facility development and upgrades. DOE could transport fuel on the Site for safety, fuel consideration, or research and development activities. In addition, DOE could undertake actions it deemed desirable, though not essential, for safety and could perform spent nuclear fuel processing, treatment, research, and development.

3.2.3.2 SRS Options 2a, 2b, and 2c. DOE analyzed three options specific to the SRS for this alternative: Option 2a deals with dry storage, Option 2b deals with wet storage, and Option 2c involves processing existing SRS aluminum-clad spent nuclear fuel and storing the remaining fuel. The amount of spent fuel that the SRS would manage includes its current inventory, as described above for Alternative 1, plus:

- 8.2 MTHM (9.0 tons) of aluminum-clad fuel
- 0.8 MTHM (0.9 tons) of stainless steel-clad fuel
- 0.3 MTHM (0.3 ton) of Zircaloy-clad fuel

Under this alternative, SRS would manage a total of about 210.8 MTHM (232.4 tons) of spent nuclear fuel. The SRS would receive spent fuel from research reactors as existing storage allowed and as new storage was constructed.

3.2.3.2.1 Option 2a - Dry Storage — Under this option, DOE would store existing SRS inventories in wet pools while developing the technology and constructing the necessary facilities to examine, characterize, and can the fuels and transfer them to a new dry storage vault to await treatment for final disposition. The SRS would proceed with the fuel rearrangement plan described above for Alternative 1 to provide acceptable storage conditions to minimize failures of the aluminum-clad material before its placement in a dry-storage container.

Placement in a dry-storage facility would require a technology development program into DOE capabilities to examine, characterize, and can aluminum-clad fuel elements before placing them in a vault. In addition, the SRS would investigate technologies for the ultimate disposition of spent nuclear fuel. In addition to a dry storage facility, the SRS would build new fuel receiving, characterization, and dry canning facilities.

3.2.3.2.2 Option 2b - Wet Storage — Under this case, DOE could rearrange existing spent nuclear fuel as described above for Alternative 1 to provide interim wet storage capacity while constructing new facilities. SRS could also modify this rearrangement plan to accept shipments of spent fuel from offsite and place them directly into the Receiving Basin for Offsite Fuels, as circumstances warrant. The new wet storage facilities required under this option would include the capability to examine and characterize fuels and to can deteriorating fuels in a stainless-steel package for placement in the new pool. DOE would move all fuel to the new storage pool once it was complete. SRS would build new fuel receiving, characterization, and wet-canning facilities as well as a new wet storage pool.

SRS would investigate technologies for the ultimate disposition of spent nuclear fuel. The SRS would build new fuel receiving, characterization, and wet-canning facilities, as well as a new wet storage pool.

3.2.3.2.3 Option 2c - Processing and Storage — Under this option, SRS would process existing aluminum-clad spent nuclear fuel to consolidate and stabilize the nuclear material for storage in vaults, and would place the stainless-steel- and Zircaloy-clad fuel and new receipts of aluminum-clad fuel in dry storage. The fuel would remain in the current wet pools while awaiting processing or the construction of new dry storage facilities. DOE would use historical F- and H-Area facilities to process the aluminum-clad fuel to safe, stable, consolidated forms.

The new facilities that the SRS would require under this option would be similar to those described for dry storage (Option 2a), except they would be much smaller because the amount of fuel to be stored would be small: only about 8.2 MTHM (9.0 tons) of aluminum and about 1.1 MTHM (1.2 tons) of nonaluminum fuel.

The SRS would investigate technologies required for the ultimate disposition of spent fuel.

3.2.4 Alternative 3 - 1992/1993 Planning Basis

3.2.4.1 Overview. This alternative assumes the continued transportation, receipt, processing, and storage of spent nuclear fuel. Foreign and university research reactor spent nuclear fuel would be sent to the INEL and the SRS. DOE would assess the construction of new facilities required to accommodate current and projected spent nuclear fuel storage requirements. This alternative would include activities related to the treatment of spent nuclear fuel, including research and development and pilot programs to support future decisions on its ultimate disposition.

3.2.4.2 SRS Options 3a, 3b, and 3c. DOE analyzed the same three options for this alternative as for Alternative 2: dry storage (Option 3a), wet storage (Option 3b), and the processing of existing SRS aluminum-clad fuel and storing the remaining fuel (Option 3c). The quantities of fuel would be somewhat greater than those for Alternative 2 because the options assume that the SRS would manage its present inventory (see Alternative 1) plus approximately:

- 11.1 MTHM (12.2 tons) of aluminum-clad fuel
- 1.3 MTHM (1.4 tons) of commercial nonaluminum-clad fuel
- 2.0 MTHM (2.2 tons) of stainless steel-clad fuel
- 0.3 MTHM (0.3 ton) of Zircaloy-clad fuel

The total spent nuclear fuel managed would equal about 216.2 MTHM (238.3 tons). The Site would receive shipments of fuel from other locations as existing space allowed and as new facilities were completed.

3.2.4.2.1 Option 3a - Dry Storage — The Site would store current inventories in existing wet pools while developing technology and constructing facilities necessary to examine, characterize, and can the fuels and transfer them to a new dry storage vault to await treatment for final disposition.

The actions that SRS would undertake under this option and the new facilities to be constructed would be the same as those described for Option 2a - Dry Storage under Alternative 2 (Decentralization) in Section 3.2.3.2.1.

3.2.4.2.2 Option 3b - Wet Storage — DOE could rearrange existing spent nuclear fuel as described in Alternative 1 above to provide interim wet storage capacity while building new facilities. The Site could also accept new shipments directly into the Receiving Basin for Offsite Fuels, as required. The actions that SRS would undertake under this option, and the new facilities to be constructed, would be the same as those described for Option 2b - Wet Storage under Alternative 2 (Decentralization) in Section 3.2.3.2.2.

3.2.4.2.3 Option 3c - Processing and Storage — Under this option, the SRS would process existing aluminum-clad spent nuclear fuel and would place the stainless steel- and Zircaloy-clad fuel and new receipts of aluminum-clad fuel in storage as described for Option 2c - Processing under Alternative 2 (Decentralization) in Section 3.2.3.2.3. The requirements for new facilities and for technology development would also be the same.

3.2.5 Alternative 4 - Regionalization

3.2.5.1 Overview. This alternative has two subalternatives. The first (Regionalization A) would involve the distribution of existing and new spent nuclear fuel among DOE sites based primarily on the similarity of fuel type, although DOE would also consider transport distances, available processing capabilities, available storage capabilities, or a combination of these factors. Under this subalternative, SRS would receive all aluminum-clad fuel and would transfer its existing inventory of stainless-steel- and Zircaloy-clad fuel to another DOE site. The SRS would manage a total of about 207.7 MTHM (228.9 tons) of spent fuel under the Regionalization A subalternative.

The second subalternative (Regionalization B) would require DOE to consolidate all existing and new spent fuel at two sites — one to the east of the Mississippi River and one to the west — depending on the location or generation site of the fuel. Under this alternative, the SRS would either receive all spent nuclear fuel in the east [approximately 246.6 MTHM (271.8 tons)] or ship its current inventory offsite to the Oak Ridge Reservation in Tennessee. An additional option if SRS becomes the Eastern Regional Site is for DOE to construct an Expended Core Facility at the SRS to manage some Naval fuel. This option is described in Appendix D of Volume 1 of this EIS.

Under either subalternative, DOE would undertake facility upgrades, replacements, and additions as appropriate. This alternative would include research and development and pilot programs to support current management and future decisions on spent fuel disposition.

3.2.5.2 SRS Options 4a, 4b, and 4c (Regionalization A). DOE analyzed three options for the regionalization of fuels by fuel type: dry storage (Option 4a), wet storage (Option 4b) and processing of existing SRS aluminum-clad fuels and storing the remaining fuel (Option 4c). This subalternative assumes that the SRS would manage:

- Its current inventory of 184.4 MTHM (203.3 tons) of aluminum-clad fuels, plus

- Approximately 23.3 MTHM (25.7 tons) of research reactor aluminum-clad fuel from other sites

The SRS would ship to the Idaho National Engineering Laboratory approximately:

- 0.6 MTHM (0.7 tons) of stainless-steel-clad fuel
- 4.6 MTHM (5.1 tons) of commercial nonaluminum-clad fuel
- 11.9 MTHM (13.1 tons) of Zircaloy-clad spent fuel

DOE would manage a total of about 207.7 MTHM (228.9 tons) of spent nuclear fuel at the SRS under this subalternative. The site would receive shipments from other locations as existing space became available and as it shipped the nonaluminum-clad fuel.

3.2.5.2.1 Option 4a - Dry Storage — The actions that the SRS would undertake under this option, and the new facilities to be constructed, would be the same as for those described for Option 2a - Dry Storage under Alternative 2 (Decentralization) in Section 3.2.3.2.1.

This option would require an extensive research and development program into capabilities to examine, characterize, and can the SRS aluminum-clad fuel for dry storage.

3.2.5.2.2 Option 4b - Wet Storage — The SRS would carry out the same actions and construct the same types of facilities under this option as it would for Option 2b - Wet Storage under Alternative 2 (Decentralization) as described in Section 3.2.3.2.2. Research and development activities would also be similar to those conducted under this Decentralization alternative, except the SRS would not perform studies on nonaluminum-clad fuels.

3.2.5.2.3 Option 4c - Processing and Storage — Under this option, the SRS would process the existing aluminum-clad fuel as described for Option 2c - under Alternative 2 (Decentralization) and place the aluminum-clad fuel received from offsite into wet storage. The requirements for new construction would be different than in Option 2c, in that dry storage facilities would not be required because the nonaluminum-clad fuels would be shipped off the site. The small amount of aluminum-clad fuel to be received could be more readily stored in pools rather than developing new dry storage. Therefore, Option 4c would require DOE to construct a new fuel receiving, wet canning and wet storage facility to manage the fuel received after the major processing operations are completed. These facilities would be much smaller than those required for other alternatives.

3.2.5.3 SRS Options 4d, 4e, 4f, and 4g (Regionalization B). DOE analyzed the same three options for the regionalization of spent fuel on the basis of geographic location as for the other alternatives: dry storage (Option 4d), wet storage (Option 4e), and processing of existing aluminum-clad fuel and storing the remaining fuel (Option 4f). In addition, it assessed the option of shipping all SRS inventory offsite (Option 4g).

The amount of material that the SRS would manage if all the spent fuel in the East were shipped to the Site would total about 246.6 MTHM (271.8 tons). This would include the current SRS inventory of about 201.5 MTHM (222.1 tons) as detailed in Section 3.2.2 plus:

- 15.2 MTHM (16.8 tons) of aluminum-clad fuel
- 28.3 MTHM (31.1 tons) of commercial nonaluminum-clad fuel
- 1.0 MTHM (1.1 ton) of stainless steel-clad fuel
- 0.6 MTHM (0.6 ton) of experimental Zircaloy-clad fuel
- less than 0.1 MTHM (0.1 ton) of other experimental fuel

The activities that DOE would have to undertake at the SRS, and the facilities that it would have to build, under the dry storage, wet storage, or processing options would be very similar to those required for the Decentralization alternative (Section 3.2.3). The differences would be that the size of the storage facilities would be somewhat greater because the amount of fuel to be managed would be larger [246.6 MTHM (271.8 tons) versus 210.8 MTHM (232.4 tons)]. In addition, DOE would conduct additional research and development on the other fuel types that SRS would manage under these options.

3.2.5.3.1 Option 4d - Dry Storage — The actions that the SRS would undertake under this option, and the new facilities to be constructed, would be similar to those described for Option 2a - Dry Storage under Alternative 2 (Decentralization) in Section 3.2.3.2.1. This option would require an extensive research and development program into capabilities to examine, characterize, and can the SRS aluminum-clad fuel for dry storage.

3.2.5.3.2 Option 4e - Wet Storage — The SRS would carry out the same actions and construct the same types of facilities under this option as it would for Option 2b - Wet Storage under Alternative 2 (Decentralization) as described in Section 3.2.3.2.2. Research and development activities would also be similar to those conducted under this Decentralization alternative.

3.2.5.3.3 Option 4f - Processing and Storage — Under this option, the SRS would process the existing aluminum-clad fuel and place nonaluminum fuel and aluminum fuel received from offsite in dry storage as described for Option 2c - Processing with storage under Alternative 2 (Decentralization). The requirements for new facilities and for research and development would also be similar.

3.2.5.3.4 Option 4g - Shipment Off the Site — Under this option, the SRS would ship its current inventory of about 201.5 MTHM (222.1 tons) to the Oak Ridge Reservation. The activities and facilities required for this option are the same as those described below for Option 5d of the Centralization alternative (Section 3.2.6.2.4).

3.2.6 Alternative 5 - Centralization

3.2.6.1 Overview. Under this alternative, DOE would collect all current and future spent nuclear fuel inventories from DOE sites, the Navy, and other sources at a single location for management until final disposition. DOE would construct new facilities at the centralized site to accommodate the increased inventories. The originating sites would characterize and stabilize their spent nuclear fuel before shipping. They would then close their spent fuel facilities. This alternative would include the centralization of activities related to the treatment of spent nuclear fuel, including research and development and pilot programs to support future decisions on its disposition.

3.2.6.2 SRS Options 5a, 5b, 5c, and 5d. DOE analyzed four options for this alternative. Three deal with shipping all DOE spent nuclear fuel to the SRS for disposition and management in dry storage (Option 5a), wet storage (Option 5b), or by processing existing aluminum-clad fuel and storing the remaining fuel (Option 5c). The fourth case involves the shipment of all SRS fuel off the Site to another location (Option 5d). Options 5a, 5b, and 5c concern the following fuels:

- 65.0 MTHM (71.6 tons) of naval fuel
- 207.7 MTHM (228.9 tons) of aluminum-clad fuel
- 2103.4 MTHM (2,318.6 tons) of Hanford defense fuel
- 27.6 MTHM (30.4 tons) of graphite fuel
- 158.8 MTHM (175.0 tons) of commercial nonaluminum-clad fuel
- 119.0 MTHM (131.2 tons) of experimental stainless-steel-clad fuel
- 77.1 MTHM (85.0 tons) of Zircaloy-clad fuel
- less than 0.1 MTHM (0.1 ton) of other fuel types

DOE would manage a total of about 2,758.6 MTHM (3,040.8 tons) of spent nuclear fuel at the SRS under the first three options. Options 5a and 5b would involve storing all the fuel on the Site. Option 5c would require processing the existing aluminum-clad fuel [184.4 MTHM (203.3 tons)] and placing the remaining nonaluminum-clad SRS fuels and all fuel received from other locations [2,574.2 MTHM (2,837.5 tons)] into dry storage. The SRS could accept shipments from offsite sources and place them in storage as it built new facilities and transferred the onsite inventory.

Under Option 5d, shipments leaving the Site would amount to about 201.5 MTHM (222.1 tons), which is equal to the inventory of spent nuclear fuel at the SRS under Alternative 1.

3.2.6.2.1 Option 5a - Dry Storage — The actions that the SRS would undertake under this option would be the same as those described for Option 2a - Dry Storage under Alternative 2 (Decentralization) in Section 3.2.3.2.1. However, the number and size of the new facilities needed to implement this centralization option would be much greater because of the larger volume of fuel that the Site would manage. In addition, DOE would have to build a new Expended Core Facility at the SRS to examine and characterize the naval fuels.

This option would require an extensive research and development program into capabilities to examine, characterize, and can SRS and other fuel types before their placement in a dry storage vault. DOE would also carry out research and development into other aspects of the management of the spent fuels, including those related to its ultimate disposition.

3.2.6.2.2 Option 5b - Wet Storage — Under this option, DOE would undertake actions similar to those described in Section 3.2.3.2.2 for Option 2b - Wet Storage under Alternative 2. As with Option 5a (Dry Storage), the SRS would have to build major new facilities to manage the large volume of fuel it would receive. DOE would also have to build a new Expended Core Facility at the SRS. Research and development would be greatly expanded as well.

3.2.6.2.3 Option 5c - Processing and Storage— DOE would process the current inventory of aluminum-clad spent fuel under this option in the same manner as described for the other alternatives. All other fuel onsite and all fuel received from elsewhere would be canned and placed in new dry storage facilities. The SRS would shut down the F- and H-Area separations facilities after processing the existing inventory of aluminum-clad fuel. Thereafter, any aluminum-clad fuel sent to the SRS would be placed in dry storage.

This option would require major new facilities, including a new Expended Core Facility. DOE would also conduct extensive research and development in spent fuel management.

3.2.6.2.4 Option 5d - Shipment Off the Site — DOE would consolidate and prepare all spent nuclear fuel on the SRS for shipment to another DOE site; this would require the construction of a new fuel characterization facility. Some fuels could require canning before shipment. SRS would use existing facilities to accomplish this. DOE would then close all SRS spent nuclear fuel-related facilities.

DOE would conduct research and development into methods of stabilizing, canning, and transporting aluminum-clad fuels, particularly that which is corroded or otherwise degraded.

3.3 Comparison of Alternatives

Table 3-3 summarizes the environmental consequences of the five alternatives. Chapter 5 presents detailed descriptions of these consequences.

In general, the levels of impacts associated with Alternatives 1 through 4 would be similar because the amounts of spent nuclear fuel that DOE would manage at the SRS under these cases would be approximately the same [e.g., about 202 to 247 MTHM (223 to 272 tons)] and activities would extend throughout the full 40-year management period. The lowest level of impact at SRS would occur under Option 4g or Option 5d (Regionalization or Centralization at another site) because DOE would ship the SRS spent fuel off the Site well before the management period ended in 2035. Alternative 5, under which DOE would ship all spent nuclear fuel to the SRS, would result in the greatest onsite impacts; the Site would have to manage approximately 2,758.6 MTHM (3,040.8 tons) of spent fuel.

Table 3-3. Comparison of impacts for the five alternatives.

ALTERNATIVE 1 - NO ACTION

	Option 1 Wet Storage
Land Use	No new facilities would be required.
Socioeconomics	No new operations jobs and only about 50 construction jobs would be created.
Cultural Resources	No new construction would be carried out. No impacts are anticipated.
Aesthetics and Scenic Resources	Facilities are in an existing industrial area not visible from public access roads or from off the Site. No impacts are anticipated. Emissions would not impact visibility.
Geology	No minerals of economic value are in affected area. No impacts are anticipated.
Air Resources	Emissions of criteria air pollutants and toxic air pollutants would be only a small fraction of air quality standards.
Water Resources	This option would not require use of additional surface water beyond the 75.7 billion liters (20 billion gallons) per year that the SRS withdraws at present. This option would not require withdrawals of additional groundwater beyond the 12.5 billion liters (3.3 billion gallons) per year the SRS uses. Activities related to this option currently use about 35.1 million liters (9.3 million gallons) of groundwater per year. Impacts would be minimal. No perennial streams or other surface waters would be affected. Accidental releases could contaminate shallow groundwater that is not a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.
Ecological Resources	Minor disturbance of wildlife due to traffic would occur. No wetlands or threatened or endangered species would be affected.
Noise	The only noise experienced by offsite populations would be generated by employee traffic and by truck and rail deliveries. There would be no change in traffic noise impacts.
Traffic and Transportation	This option would not increase site traffic. Number of LCF[f], normal transport: Worker: 6.0×10^{-4} Public: 7.0×10^{-5}
Occupational and Public Health and Safety (Radiological)	Maximum LCF[f] probabilities: Worker: 4×10^{-5} Offsite population: 4×10^{-14} (air) 1×10^{-14} (water) Annual LCF[f] incidences: Worker: 8×10^{-5} Offsite population: 2×10^{-9}

Table 3-3. (continued).

	Option 1 Wet Storage
Occupational and Public Health and Safety (Nonradiological)	Hazard index: Worker: 2 x 10^{-6} Maximally exposed individual: 2 x 10^{-7}
Utilities and Energy	Minimal changes in demand for electricity, steam, domestic water and wastewater treatment would occur. Current SRS capacities are adequate for these additions. Impacts would be minimal.
Materials and Waste Management	Annual average volume of waste generated (cubic meters)[b]: LLW: 400 TRU: 17 HLW: 0 No impact on site waste management capacities.
Accidents[c]	Greatest point estimate of risk[d]: Worker: Data not calculated[e] Colocated worker: 7.7 x 10^{-7} Maximally exposed individual: 1.6 x 10^{-7} Offsite population: 1.4 x 10^{-3}

a. Not applicable.
b. LLW = low-level waste; TRU = transuranic waste; HLW = high-level waste.
c. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.
d. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
e. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.
f. LCF = latent cancer fatalities.

Table 3-3. (continued).

ALTERNATIVE 2 - DECENTRALIZATION

	Option 2a Dry Storage	Option 2b Wet Storage	Option 2c Processing
Land Use	Most new construction would be in parts of F- and H-Areas already dedicated to industrial use. Impacts would be minimal.	Same as Option 2a.	Same as Option 2a.
Socioeconomics	Operations jobs would be filled by current employees. A maximum of about 600 construction jobs would be created.	Same as Option 2a.	Operations jobs would be filled by current employees. A maximum of about 550 construction jobs would be created.
Cultural Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Aesthetics and Scenic Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Geology	Same as Option 1.	Same as Option 1.	Same as Option 1.
Air Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Water Resources	New withdrawals of approximately 6.1 million liters (1.6 million gallons) per year of cooling water from Savannah River would be required. Impacts would be minimal.	New withdrawals of approximately 7.2 million liters (1.9 million gallons) per year of cooling water from Savannah River would be required. Impacts would be minimal.	New withdrawals of approximately 311 million liters (82.2 million gallons) per year of cooling water from Savannah River would be required. Impacts would be minimal.
	Additional groundwater withdrawals would total about 48.7 million liters (12.9 million gallons) per year. Impacts would be minimal.	Additional groundwater withdrawals would total about 50.6 million liters (13.4 million gallons) per year. Impacts would be minimal.	Same as Option 2a.
	No perennial streams or other surface waters would be affected.	No perennial streams or other surface waters would be affected.	No perennial streams or other surface waters would be affected.
	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.

Table 3-3. (continued).

	Option 2a Dry Storage	Option 2b Wet Storage	Option 2c Processing
Ecological Resources	Small increase in traffic would cause slight increase in road kills and in disturbance of wildlife due to noise. Impacts would be minimal. No wetlands or threatened or endangered species would be affected.	Same as Option 2a. Same as Option 2a.	Small increases in traffic would cause small increase in road kills and in disturbance of wildlife due to noise. Impacts would be minimal. Same as Option 2a.
Noise	Only noise experienced by communities would be generated by employee traffic and by truck and rail deliveries. Changes in traffic levels are expected to result in only very small changes in noise impacts.	Same as Option 2a.	Same as Option 2a.
Traffic and Transportation	This option would increase site traffic slightly. Number of LCF[a], normal transport: Worker: 1.0×10^{-3} Public: 1.2×10^{-4}	Same as Option 2a.	This option would increase site traffic slightly. Number of LCF[a], normal transport: Worker: 2.1×10^{-4} Public: 1.9×10^{-5}
Occupational and Public Health and Safety (Radiological)	Maximum LCF[a] probabilities: Worker: 3×10^{-5} Offsite population: 4×10^{-14} (air) 1×10^{-14} (water) Annual LCF[a] incidences: Worker: 7×10^{-5} Offsite population: 2×10^{-9}	Maximum LCF[a] probabilities: Worker: 4×10^{-5} Offsite population: 5×10^{-14} (air) 2×10^{-14} (water) Annual LCF[a] incidences: Worker: 8×10^{-5} Offsite population: 2×10^{-9}	Maximum LCF[a] probabilities: Worker: 6×10^{-5} Offsite population: 2×10^{-7} (air) 6×10^{-8} (water) Annual LCF[a] incidences: Worker: 3×10^{-2} Offsite population: 8×10^{-3}
Occupational and Public Health and Safety (Nonradiological)	Same as Option 1.	Same as Option 1.	Hazard index: Worker: 6×10^{-3} Maximally exposed individual: 5×10^{-4}
Utilities and Energy	Requirements would increase 3 to 7 percent above present levels. Current SRS capacities are adequate for these increases.	Same as Option 2a.	Very similar to Option 2a.
Materials and Waste Management	Annual average volume of waste generated (cubic meters)[b]: LLW: 400 TRU: 18 HLW: 0 No impact on site capacities.	Same as Option 2a.	Annual average volume of waste generated (cubic meters)[b]: LLW: 400 TRU: 20 HLW: 2[c] No impact on site capacities.

Table 3-3. (continued).

	Option 2a Dry Storage	Option 2b Wet Storage	Option 2c Processing
Accidents[d]	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 1.6×10^{-6} Maximally exposed individual: 3.3×10^{-7} Offsite population: 2.8×10^{-3}	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 1.7×10^{-6} Maximally exposed individual: 3.5×10^{-7} Offsite population: 3.0×10^{-3}	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 7.7×10^{-7} Maximally exposed individual: 1.6×10^{-7} Offsite population: 1.4×10^{-3}

a. NA = not applicable.
b. LLW = low-level waste; TRU = transuranic waste; HLW = high-level waste.
c. High-level waste will be generated only during approximately the first 10 years.
d. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.
e. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
f. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.
g. LCF = latent cancer fatalities.

Table 3-3. (continued).

ALTERNATIVE 3 - 1992/1993 PLANNING BASIS

	Option 3a Dry Storage	Option 3b Wet Storage	Option 3c Processing
Land Use	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Socioeconomics	Same as Option 2a.	Operations jobs would be filled by current employees. A maximum of about 650 construction jobs would be created.	Same as Option 2c.
Cultural Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Aesthetics and Scenic Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Geology	Same as Option 1.	Same as Option 1.	Same as Option 1.
Air Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Water Resources	Same as Option 2a.	Same as Option 2b.	Same as Option 2c.
Ecological Resources	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Noise	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Traffic and Transportation	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Occupational and Public Health and Safety (Radiological)	Same as Option 2a.	Same as Option 2b.	Same as Option 2c.
Occupational and Public Health and Safety (Nonradiological)	Same as Option 1.	Same as Option 1.	Same as Option 2c.
Utilities and Energy	Same as Option 2a.	Same as Option 2a.	Very similar to Option 2a.
Materials and Waste Management	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Accidents[b]	Greatest point estimate of risk[c]: Worker: Data not calculated[d] Colocated worker: 1.9×10^{-6} Maximally exposed individual: 4.0×10^{-7} Offsite population: 3.4×10^{-3}	Same as Option 3a.	Greatest point estimate of risk[c]: Worker: Data no calculated[d] Colocated worker: 1.1×10^{-6} Maximally exposed individual: 2.3×10^{-7} Offsite population: 2.0×10^{-3}

a. NA = not applicable.
b. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.
c. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
d. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.

Table 3-3. (continued).

ALTERNATIVE 4 - REGIONALIZATION A (By Fuel Type)

	Option 4a Dry Storage	Option 4b Wet Storage	Option 4c Processing
Land Use	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Socioeconomics	Same as Option 3b.	Same as Option 3b.	Same as Option 2c.
Cultural Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Aesthetics and Scenic Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Geology	Same as Option 1.	Same as Option 1.	Same as Option 1.
Air Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Water Resources	Same as Option 2a.	Same as Option 2b.	Very similar to Option 2c.
Ecological Resources	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Noise	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Traffic and Transportation	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Occupational and Public Health and Safety (Radiological)	Same as Option 2a.	Same as Option 2b.	Maximum LCF[e] probabilities: Same as Option 2c. Annual LCF[e] incidences: Worker: 3×10^{-2} Offsite population: 9×10^{-3}
Occupational and Public Health and Safety (Nonradiological)	Same as Option 1.	Same as Option 1.	Same as Option 2c.
Utilities and Energy	Very similar to Option 2a.	Same as Option 2a.	Very similar to Option 2a.
Materials and Waste Management	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.

Table 3-3. (continued).

	Option 4a Dry Storage	Option 4b Wet Storage	Option 4c Processing
Accidents[b]	Greatest point estimate of risk[c]: Worker: Data not calculated[d] Colocated worker: 2.1×10^{-6} Maximally exposed individual: 4.4×10^{-7} Offsite population: 3.7×10^{-3}	Same as Option 3a.	Greatest point estimate of risk[c]: Worker: Data not calculated[d] Colocated worker: 1.3×10^{-6} Maximally exposed individual: 2.8×10^{-7} Offsite population: 2.4×10^{-3}

a. NA = not applicable.
b. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.
c. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
d. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.
e. LCF = latent cancer fatalities.

Table 3-3. (continued).

ALTERNATIVE 4 - REGIONALIZATION B (By Location)[a]

	Option 4d Dry Storage	Option 4e Wet Storage	Option 4f Processing
Land Use	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Socioeconomics	Operations jobs would be filled by current employees. A maximum of about 700 construction jobs would be created.	Operations jobs would be filled by current employees. A maximum of about 800 construction jobs would be created.	Same as Option 3b.
Cultural Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Aesthetics and Scenic Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Geology	Same as Option 1.	Same as Option 1.	Same as Option 1.
Air Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Water Resources	Same as Option 2a.	Same as Option 2b.	Very similar to Option 2c.
Ecological Resources	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Traffic and Transportation	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.
Occupational and Public Health and Safety (Radiological)	Maximum LCF[e] incidences: Worker: 4×10^{-5} Offsite population: 5×10^{-14} (air) 2×10^{-14} (water) Annual LCF[e] incidences: Worker: 8×10^{-5} Offsite population: 2×10^{-9}	Maximum LCF[e] incidences: Worker: 5×10^{-5} Offsite population: 6×10^{-14} (air) 2×10^{-14} (water) Annual LCF[e] incidences: Worker: 1×10^{-4} Offsite population: 2×10^{-9}	Maximum LCF[e] incidences: Worker: 7×10^{-5} Offsite population: 2×10^{-7} (air) 6×10^{-8} (water) Annual LCF[e] incidences: Worker: 3×10^{-2} Offsite population: 9×10^{-3}
Occupational and Public Health and Safety (Nonradiological)	Hazard index: Worker: 2×10^{-6} Maximally exposed individual: 3×10^{-7}	Same as Option 4d.	Hazard index: Worker: 8×10^{-3} Maximally exposed individual: 6×10^{-4}
Utilities and Energy	Same as Option 2a.	Very similar to Option 2a.	Very similar to Option 2a.
Materials and Waste Management	Same as Option 2a.	Same as Option 2a.	Same as Option 2c.

Table 3-3. (continued).

	Option 4d Dry Storage	Option 4e Wet Storage	Option 4f Processing
Accidents[b]	Greatest point estimate of risk[c]: Worker: Data not calculated[d] Colocated worker: 2.0×10^{-6} Maximally exposed individual: 4.1×10^{-7} Offsite population: 3.5×10^{-3}	Same as Option 4d	Greatest point estimate of risk[c]: Worker: Data not calculated[d] Colocated worker: 1.2×10^{-6} Maximally exposed individual: 2.5×10^{-7} Offsite population: 2.1×10^{-3}

a. Impacts for Option 4g, Ship Offsite, would be the same as for Option 5d as described in the last entry in this table.
b. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.
c. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
d. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.
e. LCF = latent cancer fatalities.

Table 3-3. (continued).

ALTERNATIVE 5 - CENTRALIZATION

	Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing
Land Use	Most new construction would be in parts of F- and H-Areas already dedicated to industrial use. Additional maximum of 0.4 square kilometer (100 acres) would be converted from pine plantation to industrial use. Impacts would be minimal.	Same as Option 5a.	Same as Option 5a.
Socioeconomics	Operations jobs would be filled by present employees. A maximum of about 2,550 construction jobs would be created.	Operations jobs would be filled by present employees. A maximum of about 2,700 construction jobs would be created.	Operations jobs would be filled by present employees. A maximum of about 2,550 construction jobs would be created.
Cultural Resources	No known historical, archaeological, or paleontological resources are in areas to be affected. All areas are classified as having low or moderate probability of containing archaeological site. Impact is unlikely.	Same as Option 5a.	Same as Option 5a.
Aesthetics and Scenic Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Geology	Same as Option 1.	Same as Option 1.	Same as Option 1.
Air Resources	Same as Option 1.	Same as Option 1.	Same as Option 1.
Water Resources	Same as Option 2a.	Same as Option 2b.	Same as Option 2c.
	Additional groundwater withdrawals would total about 67.7 million liters (17.9 million gallons) per year. Impacts would be minimal.	Additional groundwater withdrawals would total about 69.6 million liters (18.4 million gallons) per year. Impacts would be minimal.	Same as Option 5a.
	No perennial streams or other surface waters would be affected.	Same as Option 5a.	Same as Option 5a.
	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.	Accidental releases could contaminate shallow groundwater that is not used as a source for drinking water or domestic use. Releases would not affect surface streams or drinking water aquifers.

Table 3-3. (continued).

	Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing
Ecological Resources	Same as Option 2a, plus Loss of up to 0.4 square kilometer (100 acres) of loblolly pine. Impacts would be minor.	Same as Option 5a.	Same as Option 5a, plus Increased disturbance due to more worker traffic. Impacts would be minor.
Noise	Same as Option 2a.	Same as Option 2a.	Same as Option 2a.
Traffic and Transportation	Same as Option 2a.	This option would increase site traffic by about 17 percent. Impacts would be small. Number of LCFs[a] would be same as for Option 2b for normal transport.	Same as Option 2c.
Occupational and Public Health and Safety (Radiological)	Maximum LCF[a] probabilities: Worker: 4×10^{-4} Offsite population: 5×10^{-13} (air) 2×10^{-13} (water) Annual LCF[a] incidences: Worker: 9×10^{-4} Offsite population: 2×10^{-8}	Maximum LCF[a] probabilities: Worker: 5×10^{-4} Offsite population: 6×10^{-13} (air) 2×10^{-13} (water) Annual LCF[a] incidences: Worker: 1×10^{-3} Offsite population: 3×10^{-8}	Maximum LCF[a] probabilities: Worker: 6×10^{-4} Offsite population: 2×10^{-7} (air) 6×10^{-8} (water) Annual LCF[a] incidences: Worker: 3×10^{-2} Offsite population: 9×10^{-3}
Occupational and Public Health and Safety (Nonradiological)	Same as Option 1.	Same as Option 1.	Same as Option 2c.
Utilities and Energy	Similar to Option 2a.	Similar to Option 2a.	Requirements for electricity would increase by about 17 percent. Other increases would be similar to Option 2c. Impacts would be minor.
Materials and Waste Management	Same as Option 2a.	Annual average volume of waste generated (cubic meters)[b]: LLW: 500 TRU: 20 HLW: 0 No impact on site capacities.	Annual average volume of waste generated (cubic meters)[b]: LLW: 1,700 TRU: 20 HLW: 2[c] No impact on site capacities.

Table 3-3. (continued).

	Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing
Accidents[d]	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 4.0×10^{-6} Maximally exposed individual: 8.4×10^{-7} Offsite population: 7.2×10^{-3}	Same as Option 5a.	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated worker: 3.3×10^{-6} Maximally exposed individual: 6.8×10^{-7} Offsite population: 5.8×10^{-3}

a. NA = not applicable.
b. LLW = low-level waste; TRU = transuranic waste; HLW = high-level waste.
c. High-level waste will be generated only during approximately the first 10 years.
d. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.
e. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
f. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.
g. LCF = latent cancer fatalities.

Table 3-3. (continued).

ALTERNATIVE 5 - CENTRALIZATION
ALTERNATIVE 4 - REGIONALIZATION B

	Option 4g and Option 5d[b] Ship Out
Land Use	Same as Option 1.
Socioeconomics	No new operations jobs and only about 200 construction jobs would be created.
Cultural Resources	Same as Option 1.
Aesthetics and Scenic Resources	Same as Option 1.
Geology	Same as Option 1.
Air Resources	Same as Option 1.
Water Resources	This option would require new withdrawals of approximately 3.0 million liters (790 thousand gallons) per year of cooling water from the Savannah River. Impacts would be minimal. It also would require additional groundwater withdrawals of about 38.1 million liters (10.1 million gallons) per year. Impacts would be minimal. Impacts to surface water and groundwater would be similar to those from Option 1.
Ecological Resources	Same as Option 1.
Noise	Same as Option 2a.
Traffic and Transportation	NA[a]
Occupational and Public Health and Safety (Radiological)	Less than Option 1.
Occupational and Public Health and Safety (Nonradiological)	Same as Option 1.
Utilities and Energy	Requirements would increase 2 to 6 percent above current levels during first 10 years. Current SRS capacities are adequate for these increases.
Materials and Waste Management	Annual average volume of waste generated initial 10 years only (cubic meters)[c]: LLW: 400 TRU: 5 HLW: 0

Table 3-3. (continued).

	Option 4g and Option 5d[b] Ship Out
Accidents[d]	Greatest point estimate of risk[e]: Worker: Data not calculated[f] Colocated Worker: Option 4g: 8.1×10^{-7} Option 5d: 8.2×10^{-7} Maximally exposed individual: Option 4g: 1.7×10^{-7} Option 5d: 1.7×10^{-7} Offsite population: Option 4g: 1.4×10^{-3} Option 5d: 1.4×10^{-3}

a. NA = not applicable.
b. Impacts for Option 4g (Regionalization-B) are the same as for Option 5d.
c. LLW = low-level waste; TRU = transuranic waste; HLW = high-level waste.
d. Data is provided as adjusted point estimates of risk by receptor group to demonstrate a relative comparison of each alternative on an option-by-option basis. The adjusted values were taken from Tables 5-27 through 5-29.
e. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
f. The safety analysis reports from which information was extracted were written before issuance of DOE Order 5480.23; previous orders did not require the inclusion of workers.

4. AFFECTED ENVIRONMENT

4.1 Overview

This section describes the existing environment at the Savannah River Site (SRS) and nearby areas. Its purpose is to support the assessment of environmental consequences of the alternative actions regarding spent nuclear fuels described in Chapter 3. Chapter 5 describes the environmental consequences in detail.

4.2 Land Use

The SRS occupies an area of approximately 198,000 acres (800 square kilometers) in western South Carolina, in a generally rural area about 25 miles (40 kilometers) southeast of Augusta, Georgia. The SRS, which is bordered by the Savannah River to the southwest, includes portions of Aiken, Barnwell, and Allendale Counties (Figure 2-1).

Land use on the SRS falls into three major categories: forest/undeveloped, water/wetlands, and developed facilities. About 181,500 acres (735 square kilometers) of the SRS area are undeveloped (USDA 1991a). Approximately 90 percent of this undeveloped area is forested (Cummins et al. 1991). In 1952, an interagency agreement between the U.S. Department of Energy [DOE, which was then the Atomic Energy Commission (AEC)] and the Forest Service, U.S. Department of Agriculture, created an SRS forest management program. In 1972, the AEC designated the SRS as a National Environmental Research Park (NERP); at present, approximately 14,000 acres (57 square kilometers or 7 percent) of the SRS area are designated as "Set-Asides," areas specifically protected for environmental research activities that are coordinated either through the University of Georgia Savannah River Ecology Laboratory (SREL) or the Savannah River Technology Center (SRTC; Davis 1994). Administrative, production, and support facilities occupy approximately 5 percent of the total SRS land area.

DOE is considering decisions that could affect the long-range land use of the SRS. Programmatic decisions on the reconfiguration of the nuclear weapons complex, spent nuclear fuel interim strategies, and waste management and environmental restoration activities that could result in significant changes in the SRS mission are in the early stages of discussion. In the shorter term, however, a Land Use Technical Committee consisting of representatives from DOE, Westinghouse

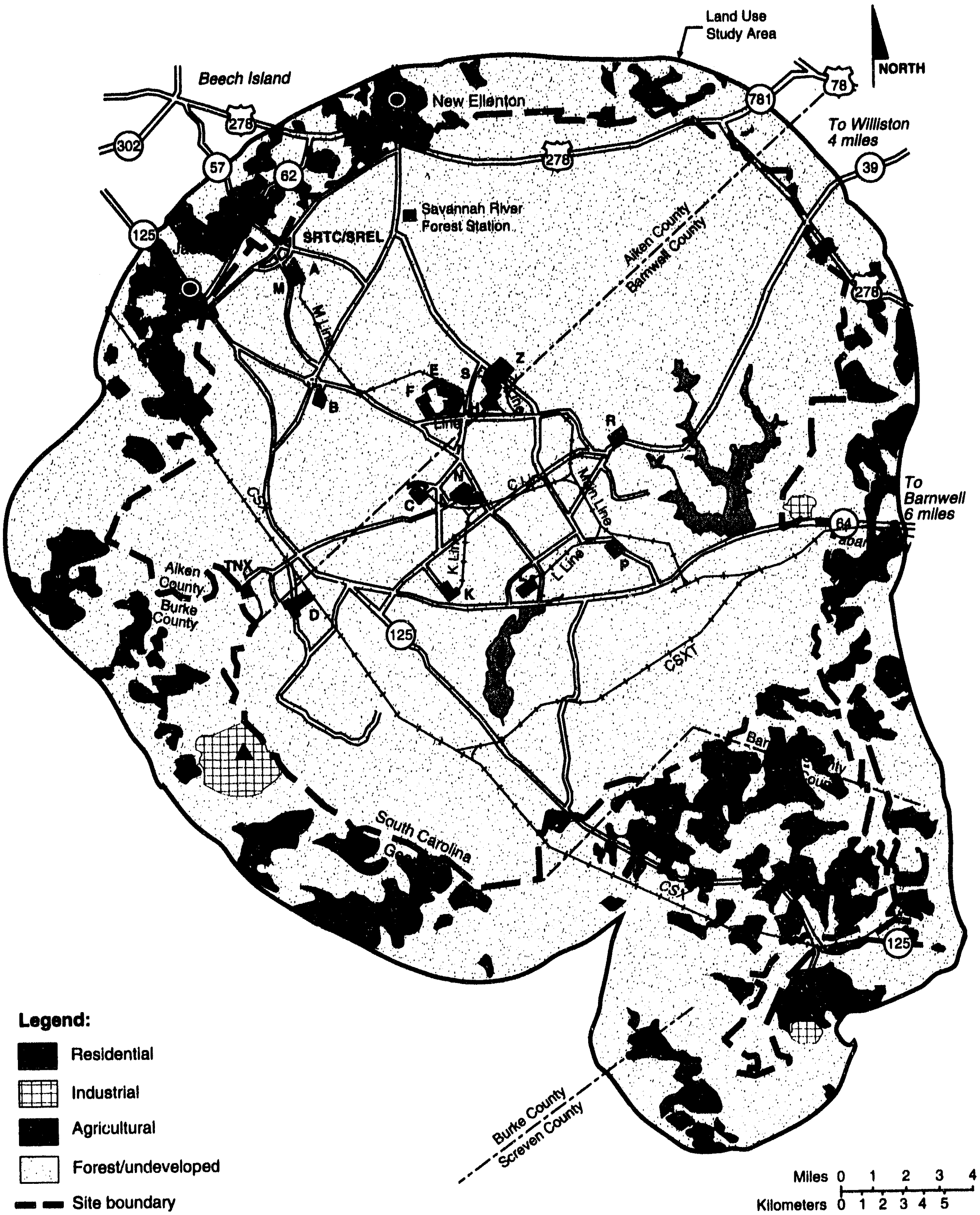
Savannah River Company, and various stakeholder groups is evaluating alternative land use strategies and potential future uses. These activities are consistent with the guidelines for land use plans contained in DOE Order 4320.1B, "Site Development Planning," and in the Resource Conservation and Recovery Act (RCRA) and the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA).

Land use bordering SRS is primarily forest and agricultural. There is also a significant amount of open water and nonforested wetlands along the Savannah River valley. Incorporated and industrial areas are the only other significant use of land in the vicinity (Figure 4-1). None of the three counties in which the SRS is located has zoned any of the Site land. The only adjacent area with any zoning is the Town of New Ellenton, which has two zoning categories for lands that bound SRS - urban development and residential development. The closest residences to the SRS boundary include several within 200 feet (61 meters) of the Site perimeter to the west, north, and northeast.

Various industrial, manufacturing, medical, and farming operations are conducted in areas surrounding the Site. Major industrial and manufacturing facilities in the area include textile mills, plants producing polystyrene foam and paper products, chemical processing plants, and a commercial nuclear power plant. Farming is diversified in the region and includes crops such as peaches, watermelon, cotton, soybeans, corn, and small grains.

There is a wide variety of public outdoor recreation facilities in the SRS region (Figure 4-2). Federal outdoor recreation facilities include portions of the Sumter National Forest [47 miles (75 kilometers) to the northwest of the Site], the Santee National Wildlife Refuge [50 miles (80 kilometers) to the east], and the Clarks Hill/Strom Thurmond Reservoir, a U.S. Army Corps of Engineers impoundment [43 miles (70 kilometers) to the northwest]. There are also a number of state, county, and local parks in the region, most notably Redcliffe Plantation, Rivers Bridge, Barnwell and Aiken County State Parks in South Carolina, and Mistletoe State Park in Georgia (HNUS 1992a).

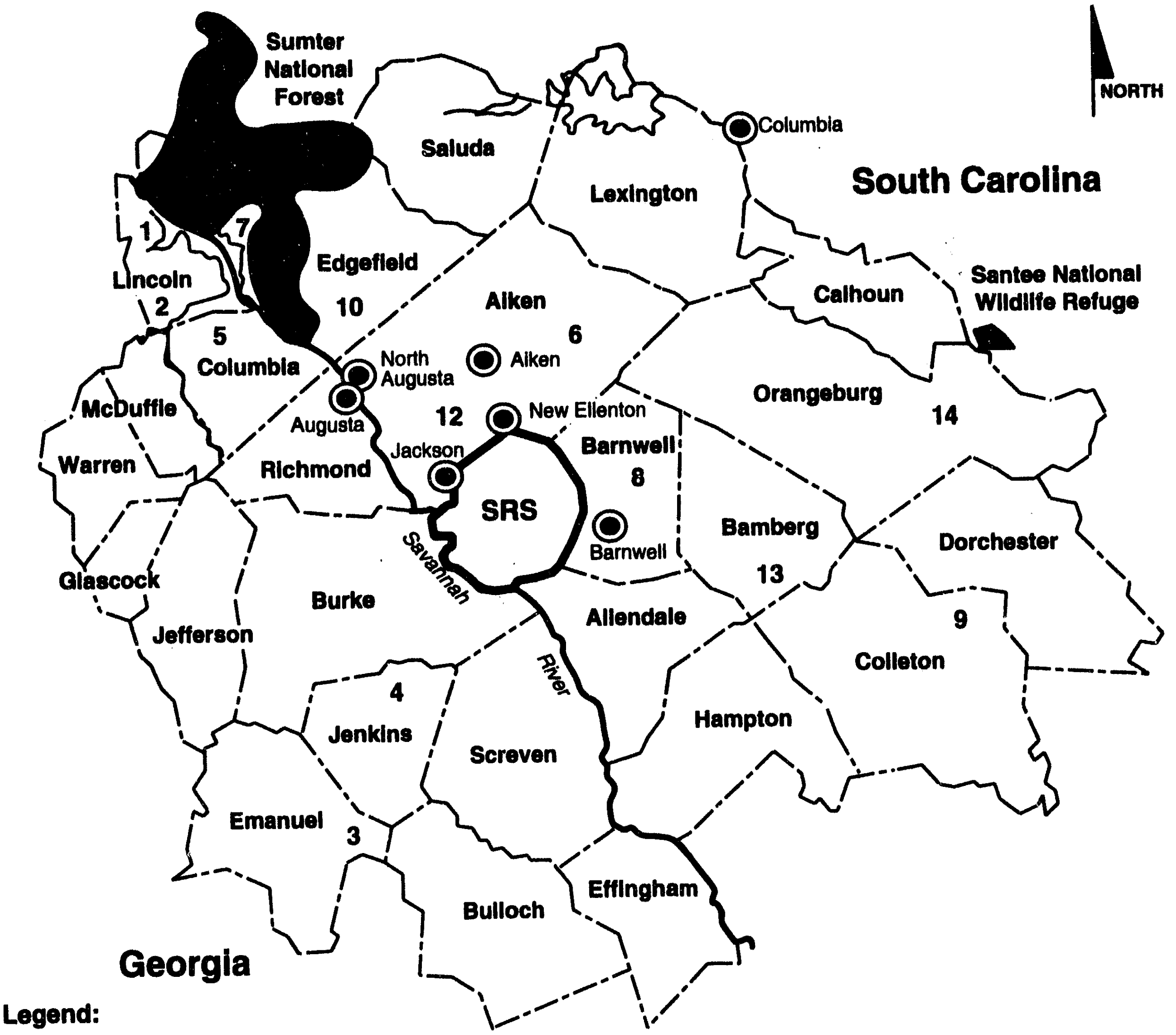
The SRS is a controlled area with public access limited to through traffic on South Carolina Highway 125 (SRS Road A), U.S. Highway 278, SRS Road 1, and the CSX railway. The SRS does not contain any public recreation facilities. However, the SRS conducts controlled deer hunts each fall, from mid-October through mid-December; hunters can also kill feral hogs during these hunts.



Source: DOE (1993a)

SFIG 0401

Figure 4-1. Generalized land use at the Savannah River Site and vicinity.



Legend:

Georgia

1 Clarks Hill Reservoir
2 Elijah Clark State Park
3 George L. Smith State Park
4 Magnolia Springs State Park
5 Mistletoe State Park

South Carolina

6 Aiken State Park
7 Baker Creek State Park
8 Barnwell State Park
9 Colleton State Park
10 Hamilton Branch State Park
11 Hickory Knob State Park
12 Redcliffe Plantation State Park
13 Rivers Bridge State Park
14 Santee State Park

Miles 0 10 20 30 40
Kilometers 0 10 20 30 40 50 60

Source: HNUS (1992a)

SFIG 0402

Figure 4-2. Federal and state forests and parks within a 2-hour drive from Savannah River Site.

The intent of the hunts is to control the resident populations of these animals and to reduce animal-vehicle accidents on SRS roads.

No onsite areas are subject to Native American treaty rights. The SRS does not contain any prime farmland.

4.3 Socioeconomics

This section discusses baseline socioeconomic conditions within a region of influence where approximately 90 percent of the SRS workforce lived in 1992. The SRS region of influence includes Aiken, Allendale, Bamberg, and Barnwell Counties in South Carolina, and Columbia and Richmond Counties in Georgia (Figure 4-2).

4.3.1 Employment and Labor Force

The labor force living in the region of influence increased from about 150,550 to 209,000 between 1980 and 1990. In 1990, approximately 75 percent of the total labor force in the region of influence lived in Richmond and Aiken counties. Assuming a constant unemployment rate of 5.8 percent, the regional labor force is likely to increase to approximately 257,000 by 1995 (Table 4-1).

Between 1980 and 1990, total employment in the region of influence increased from 139,504 to 199,161, an average annual growth rate of approximately 5 percent. Table 4-1 lists projected employment data for the six-county region of influence. As shown, by 1995 employment levels should increase 22 percent to approximately 242,000. The unemployment rates for 1980 and 1990 were 7.3 percent and 4.7 percent, respectively (HNUS 1992a).

In 1990, employment at the SRS was 20,230 (DOE 1993a), representing 10 percent of the employment in the region of influence. In Fiscal Year 1992, employment at the SRS increased approximately 15 percent to 23,351, with an associated payroll of more than $1.1 billion. As shown in Table 4-1, Site employment should decrease to approximately 20,000 by 1995 (Turner 1994).

Table 4-1. Forecast employment and population data for the Savannah River Site and the region of influence.[a]

Year	Labor Force (Region)	Employment (Region)	SRS Employment[b]	Population (Region)
1994	254,549	239,785	21,528	456,892
1995	256,935	242,033	20,055	461,705
1996	258,500	243,507	19,262	465,563
1997	260,680	245,561	18,923	468,665
1998	263,121	247,860	18,809	471,176
1999	265,694	250,284	19,036	473,186
2000	268,430	252,861	18,695	474,820
2001	271,265	255,532	18,695	476,179
2002	274,238	258,332	18,695	477,332
2003	277,318	261,234	18,695	478,340
2004	280,415	264,151	18,695	479,182

a. Source: HNUS (1993).
b. Turner (1994).

4.3.2 Personal Income

Personal income in the six-county region has doubled during the past two decades, increasing from approximately $3.4 billion in 1970 to almost $6.9 billion by 1989 (in constant 1991 dollars). Together, Richmond and Aiken Counties accounted for 75.4 percent of the personal income in the region of influence in 1989, because these two counties provide most of the employment opportunities in the region. Personal income in the region is likely to increase 3 percent to approximately $7.1 billion by 1995 and to almost $8.2 billion by 2000 (HNUS 1992a).

4.3.3 Population

Between 1980 and 1990, the population in the region of influence increased 13 percent from 376,058 to 425,607. More than 88 percent of the 1990 population lived in Aiken (28.4 percent), Columbia (15.5 percent), and Richmond (44.6 percent) Counties. Table 4-1 also lists population data for the region of influence forecast to 2004. According to census data, in 1990 the estimated average number of persons per household in the six-county region was 2.72, and the median age of the population was 31.2 years (HNUS 1992a).

4.3.4 Housing

From 1980 to 1990, the number of year-round housing units in the six-county region increased 23.2 percent from 135,866 to 167,356. In 1990, approximately 68 percent of the total housing units were single-family units, 18 percent were multifamily units, and 14 percent were mobile homes. In the same year, the region had a 4.7-percent vacancy rate with 7,818 available unoccupied housing units. Of the available unoccupied units, 29 percent (2,267) were available for sale and 71 percent (5,551) were available for rent (HNUS 1992a).

4.3.5 Community Infrastructure and Services

Public education facilities in the six-county region include 95 elementary and intermediate schools and 25 high schools. Aside from the public school systems, 42 private schools and 16 post-secondary facilities are available to residents in the region (HNUS 1992a).

Based on a combined average daily attendance for elementary and high school students in the region of influence in 1988, the average number of students per teacher was 16. The highest ratio was in Columbia County high schools where there were 19 students per teacher (1987-1988). The lowest ratio occurred in Barnwell County's District 29 high school, which had only 12 students per teacher (1988-1989) (HNUS 1992a).

The six-county region has 14 major public sewage treatment facilities with a combined design capacity of 302.2 million liters (79.8 million gallons) per day. In 1989, these systems were operating at approximately 56 percent of capacity, with an average daily flow of 170 million liters (44.9 million gallons) per day. Capacity utilization ranged from 45 percent in Aiken County to 80 percent in Barnwell County (HNUS 1992a).

There are approximately 120 public water systems in the region of influence. About 40 of these county and municipal systems are major facilities, while the remainder serve individual subdivisions, water districts, trailer parks, and miscellaneous facilities. In 1989, the 40 major facilities had a combined total capacity of 576.3 million liters (152.2 million gallons) per day. With an average daily flow rate of approximately 268.8 million liters (71 million gallons) per day, these systems were operating at 47 percent of total capacity in 1989. Facility utilization rates ranged from 13 percent in Allendale County to 84 percent in the City of Aiken (HNUS 1992a).

Eight general hospitals operate in the six-county region with a combined bed capacity in 1987 of 2,433 (5.7 beds per 1,000 population). Four of the eight general hospitals are in Richmond County; Aiken, Allendale, Bamberg, and Barnwell Counties each have one general hospital. Columbia County has no hospital. In 1989, there were approximately 1,295 physicians serving the regional population, which represents a physician-to-population ratio of 3 to 1,000. This ratio ranged from 0.8 physician per 1,000 people in Aiken and Allendale Counties to 5.4 physicians per 1,000 people in Richmond County (HNUS 1992a).

Fifty-six fire departments provide fire protection services in the region of influence. Twenty-seven of these are classified as municipal fire departments, but many provide protection to rural areas outside municipal limits. The average number of firefighters in the region in 1988 was 3.8 per 1,000 people, ranging from 1.6 per 1,000 in Richmond County to 10.2 per 1,000 in Barnwell County (HNUS 1992a).

The county sheriff departments and municipal police departments provide most law enforcement services in the region of influence. In addition, state law enforcement agents and state troopers assigned to each county provide protection and assist county and municipal law enforcement officers. In 1988, the average ratio in the region of full-time police officers employed by state, county, and local agencies per 1,000 population was 2.0. This ratio ranged from 1.4 per 1,000 in Columbia County to 2.5 per 1,000 in Richmond County (HNUS 1992a).

4.3.6 Government Fiscal Structure

This section discusses the fiscal structure of Aiken and Barnwell Counties because these two counties would have the greatest potential for fiscal impacts from changes at SRS.

Public services provided by Aiken County are funded principally through the county's general fund. In Fiscal Year 1988, revenues and expenditures of this fund were $15.5 million and $18 million, respectively. The current property tax rate is 55.8 mills for county operations and 8.0 mills for debt service. Long-term general obligation bond indebtedness was $9.3 million at the end of Fiscal Year 1988, and reserve general obligation bond indebtedness was $5.5 million. The assessed value of property in the county was $182.5 million in Fiscal Year 1988 (HNUS 1992a).

Assuming revenues and expenditures increase in proportion to projected growth in the employment and population, estimated revenues and expenditures for Aiken County over the period from Fiscal Year 1990 to Fiscal Year 2000 will be $15.6 million to $17.0 million (in constant 1988 dollars) (HNUS 1992a).

Public services provided by Barnwell County also are funded principally through the county's general fund. In Fiscal Year 1988, revenues and expenditures of this fund were $4.0 million and $4.9 million, respectively. The property tax rate is 23.9 mills of assessed valuation. Budgeted Fiscal Year 1990 revenues were approximately $4.5 million (HNUS 1992a).

4.4 Cultural Resources

4.4.1 Archeological Sites and Historic Structures

Field studies conducted under an ongoing program over the past two decades by the South Carolina Institute of Archeology of the University of South Carolina, under contract to DOE and in consultation with the South Carolina State Historic Preservation Officer, have provided considerable information about the distribution and content of archeological and historic resources on the SRS. By the end of Fiscal Year 1992, approximately 60 percent of the Site had been examined, and 858 archeological (historic and prehistoric) sites had been identified; these include 706 prehistoric and 350 historic components, some of which are mixed (i.e., contain elements of both). Of the 858 sites, 53 have been determined to be eligible for the National Register of Historic Places; 650 have not been evaluated. Approximately 21 of the 53 (40 percent) are historic sites, such as building foundations; none are standing structures. These sites provide knowledge of the area's history before 1820. The remainder are primarily prehistoric sites and some are mixed (historic and prehistoric). No SRS facilities have been nominated for eligibility to the National Register for Historic Places and there are no plans for such a nomination at this time (Brooks 1993; Brooks 1994). The existing SRS nuclear production facilities are not likely to be eligible for the National Register, either because they might lack architectural integrity, might not represent a particular architectural style, or might not contribute to the broad historic theme of the Manhattan Project and initial nuclear materials production (DOE 1993a).

Archeologists have divided areas of the SRS into three sensitivity zones related to their potential for containing sites with multiple archeological components or dense or diverse artifacts, and their potential for eligibility to the National Register of Historic Places (SRARP 1989).

- Zone 1 is the zone of the highest archeological site density with a high probability of encountering large archeological sites with dense and diverse artifacts, and high potential for nomination to the National Register of Historic Places.

- Zone 2 covers areas of moderate archeological site density that should contain sites of similar composition. Activities in this zone have a moderate probability of encountering archeological sites, but a low probability of encountering large sites with more than three prehistoric components. All areas within the zone are conducive to site preservation. The zone has moderate potential for encountering sites that would be eligible for nomination to the National Register of Historic Places.

- Zone 3 covers areas of low archeological site density. Activities in this zone have a low probability of encountering archeological sites and virtually no chance of encountering large sites with more than three prehistoric components; potential for site preservation is low. Some exceptions to this definition have been discovered in Zone 3, so some sites in the zone could be considered eligible for nomination to the National Register of Historic Places.

4.4.2 Native American Cultural Resources

In conjunction with 1991 studies related to a proposed New Production Reactor, DOE conducted an investigation of Native American concerns over religious rights in the Central Savannah River Valley. During this study three Native American groups - the Yuchi Tribal Organization, the National Council of Muskogee Creek, and the Indian People's Muskogee Tribal Town Confederacy - expressed concerns over sites and items of religious significance on the SRS. DOE has included these organizations on its environmental mailing list and sends them documents about SRS environmental activities (NUS 1991a).

Native American resources in the region include villages or townsites, ceremonial lodges, burial sites, cemeteries, and areas containing traditional plants for certain rituals. Villages or townsites might contain a variety of sensitive features associated with different ceremonies and rituals. The Yuchi and

Muskogee Creek tribes have expressed concerns that the area might contain several plants traditionally used in tribal ceremonies (DOE 1993a).

4.4.3 Paleontological Resources

Invertebrate fossil remains occur within the McBean, Barnwell, and Congaree formations of the Eocene Age (54 million to 39 million years ago) on the SRS. Relatively large quantities of marine invertebrate fossils have been recorded for the McBean and Barnwell Formations. Relative assessment of fossil localities is difficult because the South Carolina Geological Survey has not established criteria for, or registry of, important paleontological locations (DOE 1991b).

4.5 Aesthetics and Scenic Resources

The dominant aesthetic setting in the vicinity of the SRS consists mainly of agricultural land and forest, with some limited residential and industrial areas. Because of the distance to the Site boundary, the rolling terrain, normally hazy atmospheric conditions, and heavy vegetation, SRS facilities are not generally visible from off the Site. The few locations that have views of some of the SRS structures are quite distant from the facility [5 miles (8 kilometers) or more].

SRS land is heavily wooded, and developed areas occupy only approximately 5 percent of the total land area. The facilities are scattered across the SRS and are brightly lit at night. Typically, the reactors and principal processing facilities are large concrete structures as much as 100 feet (30 meters) high and usually colocated with lower administrative and support buildings and parking lots. A 500-foot cooling tower is located in K-Area. The facilities are visible in the direct line-of-sight when approaching them from SRS access roads. Otherwise, heavily wooded areas that border the SRS road system and public highways that cross the Site limit views of the facilities.

4.6 Geology

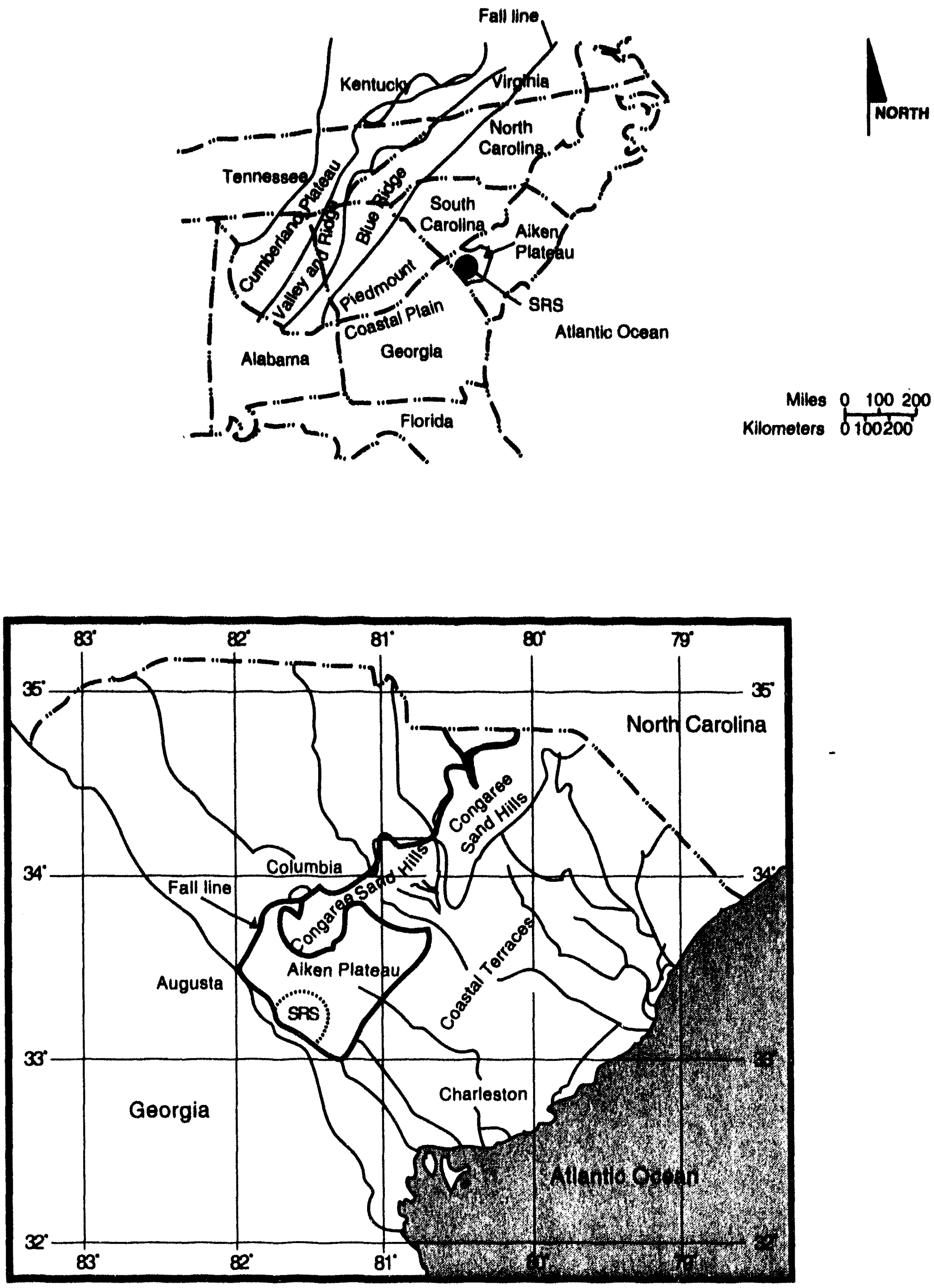
The SRS is on the Upper Atlantic Coastal Plain of South Carolina, which consists of 213 to 366 meters (700 to 1,200 feet) of sands, clays, and limestones of Tertiary and Cretaceous age. These sediments are underlain by sandstones of Triassic age and older metamorphic and igneous rocks (Arnett et al. 1993). There are no known capable faults on the SRS or volcanic activities within 800 kilometers (500 miles) of the Site.

4.6.1 General Geology

The SRS is in the Coastal Plain physiographic province of western South Carolina, approximately 32 kilometers (20 miles) southeast of the Fall Line, which separates the Piedmont and Coastal Plain provinces (Figure 4-3). The Coastal Plain province is underlain by a wedge of seaward-dipping and thickening unconsolidated and semiconsolidated sediments that extend from the Fall Line to the Continental Shelf (Figure 4-4).

In South Carolina, the Coastal Plain province is divided into the Upper Coastal Plain and the Lower Coastal Plain. Subdivisions of the Coastal Plain in the State include the Aiken Plateau and the Congaree Sand Hills in the Upper Coastal Plain, and the Coastal Terraces in the Lower Coastal Plain. The Congaree Sand Hills trend along the Fall Line northeast and north of the Aiken Plateau. The Savannah and Congaree Rivers bound the Aiken Plateau, on which the SRS is located; the plateau extends from the Fall Line to the Coastal Terraces. The surface of the plateau is highly dissected and characterized by broad interfluvial areas with narrow steep-sided valleys. The plateau is generally well drained, although poorly drained depressions (Carolina bays) do exist (DOE 1991b). Because of the proximity of the SRS to the Piedmont province, it has more relief than areas that are nearer to the coast, with onsite elevations ranging from 27 to 128 meters (89 to 420 feet) above mean sea level.

The sediments of the Atlantic Coastal Plain of South Carolina overlie a basement complex composed of Paleozoic crystalline and Triassic sedimentary rocks. These sediments dip gently seaward from the Fall Line and range in age from Late Cretaceous to Recent. The sedimentary sequence thickens from essentially zero at the Fall Line to more than 1,219 meters (4,000 feet) at the coast. Regional dip is to the southeast. Coastal Plain sediments underlying the SRS consist of sandy clays and clayey sands, although occasional beds of clean sand, gravel, clay, or carbonate occur (Figure 4-5). Two clastic limestone zones occur within the Tertiary age sequence. These calcareous zones vary in thickness from about 0.6 meter (2 feet) to approximately 24 meters (80 feet). Most of the clastic sediments are unconsolidated, but thin semiconsolidated beds also occur (DOE 1991b). Underlying sediments are dense crystalline igneous and metamorphic rock or younger consolidated sediments of the Triassic Period. The Triassic formations and older igneous and metamorphic rocks are separated hydrologically from the overlying Coastal Plain sediments by a regional aquitard, the



Source: DOE (1991a)

SFIG 0403

Figure 4-3. Location of the Savannah River Site in the southern United States.

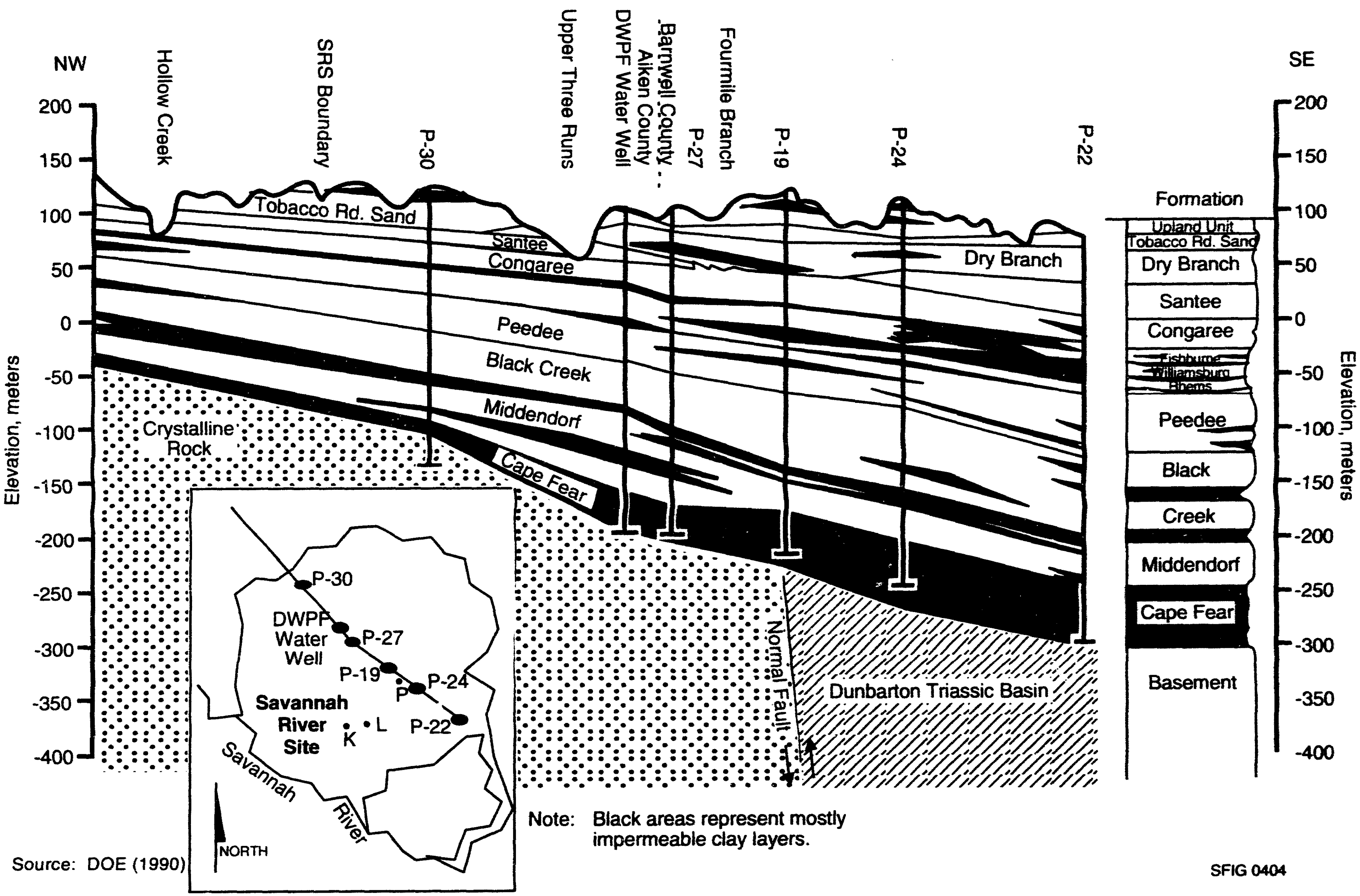


Figure 4-4. Generalized subsurface cross-section across the Savannah River Site.

Age		Unit		Lithology
Tertiary	Miocene?		"upland unit"	Clayey, silty sands, conglomerates, pebbly sands, and clays; clay clasts common
Tertiary	Eocene	Barnwell Group	Tobacco Road Sand	Red, purple, and orange, poorly to well-sorted sand and clayey sand with abundant clay laminae
Tertiary	Eocene	Barnwell Group	Dry Branch Fm.	Tan, yellow, and orange, poorly to well-sorted sand with tan and gray clay layers near base; calcareous sands and clays and limestone in lower part downdip
Tertiary	Eocene	Barnwell Group	Clinchfield Fm.	Biomoldic limestone, calcareous sand and clay, and tan and yellow sand
Tertiary	Eocene	Orangeburg Group	Santee Ls. / Warley-Hill Fm.	Micritic, calcarenitic, shelly limestone, and calcareous sands; interbedded yellow and tan sands and clays; green clay and glauconitic sand near base
Tertiary	Eocene	Orangeburg Group	Congaree Fm.	Yellow, orange, tan, and greenish-gray, fine to coarse, well-sorted sand; thin clay laminae common
Tertiary	Paleocene	Black Mingo Group	Fishburne Fm. / Williamsburg Fm.	Light gray, silty sand interbedded with gray clay
Tertiary	Paleocene	Black Mingo Group	Ellenton Formation	Black and gray, lignitic, pyritic sand and interbedded clays with silt and sand laminae
Upper Cretaceous		Lumbee Group	Peedee Formation	Gray and tan, slightly to moderately clayey sand; gray red, purple, and orange clays common in upper part
Upper Cretaceous		Lumbee Group	Black Creek Formation	Tan and light to dark gray sand; dark clays common in middle and oxidized clays at top
Upper Cretaceous		Lumbee Group	Middendorf Formation	Tan and gray, slightly to moderately clayey sand; gray red, and purple clays near top
Upper Cretaceous		Lumbee Group	Cape Fear Formation	Gray, clayey sand with some conglomerates, and sandy clay; moderately to well indurated

Paleozoic Crystalline Basement
or Triassic Newark Supergroup

SFIG 0405

Figure 4-5. Stratigraphy of the SRS region.

Appleton Confining System (Arnett et al. 1993). Section 4.8.2 contains a detailed discussion of hydrogeology on the SRS.

4.6.2 Geologic Resources

SRS construction activities have used clay, sand, and gravel to a limited extent. These materials are not of major economic value due to their abundance throughout the region. The SRS historically has been a major user of groundwater in the region, withdrawing about 33 million liters (9 million gallons) per day. Section 4.8.2 describes the groundwater resources at the SRS.

4.6.3 Seismic and Volcanic Hazards

The closest offsite fault system of significance is the Augusta Fault Zone, approximately 40 kilometers (25 miles) from the SRS. In this fault zone, the Belair Fault has experienced the most recent movement, but it is not considered capable of generating major earthquakes (DOE 1987a). There is no conclusive evidence of recent displacement along any fault within 320 kilometers (200 miles) of the SRS, with the possible exception of the buried faults in the epicentral area of the 1886 earthquake at Charleston, South Carolina, approximately 145 kilometers (90 miles) away (DOE 1991b). Faulting in the subsurface Coastal Plain sediments in the Charleston vicinity has been suggested, based on structure contour mapping of the Eocene-Oligocene unconformity, which lies at a depth of about 30 to 61 meters (100 to 200 feet) below ground surface (WSRC 1993b). However, because it is not known if these faults offset sediments younger than Eocene-Oligocene, these shallow faults cannot be related to modern earthquakes that occur at depths greater than about 1.9 kilometers (1.2 miles). Figure 4-6 shows the geologic structures within 150 kilometers (95 miles) from the SRS, some of which are discussed above.

Several Triassic-Jurassic basins, 140 to 230 million years old, have been identified in the Coastal Plain province of South Carolina and Georgia. The Dunbarton Triassic basin, which underlies a portion of the SRS, was formed by fault movement resulting from extensional forces operating during the formation of the Atlantic Ocean. After the erosion of basin margins and infilling of the basin with Triassic age sediments, possible movement of an opposite sense to that during basin formation occurred along the fault during the Late Cretaceous age. Geophysical data indicate minimal movement on faults at the basement-Coastal Plain interface, with the exception of possible reverse fault motion along the Pen Branch Fault up into the Tertiary (WSRC 1993b).

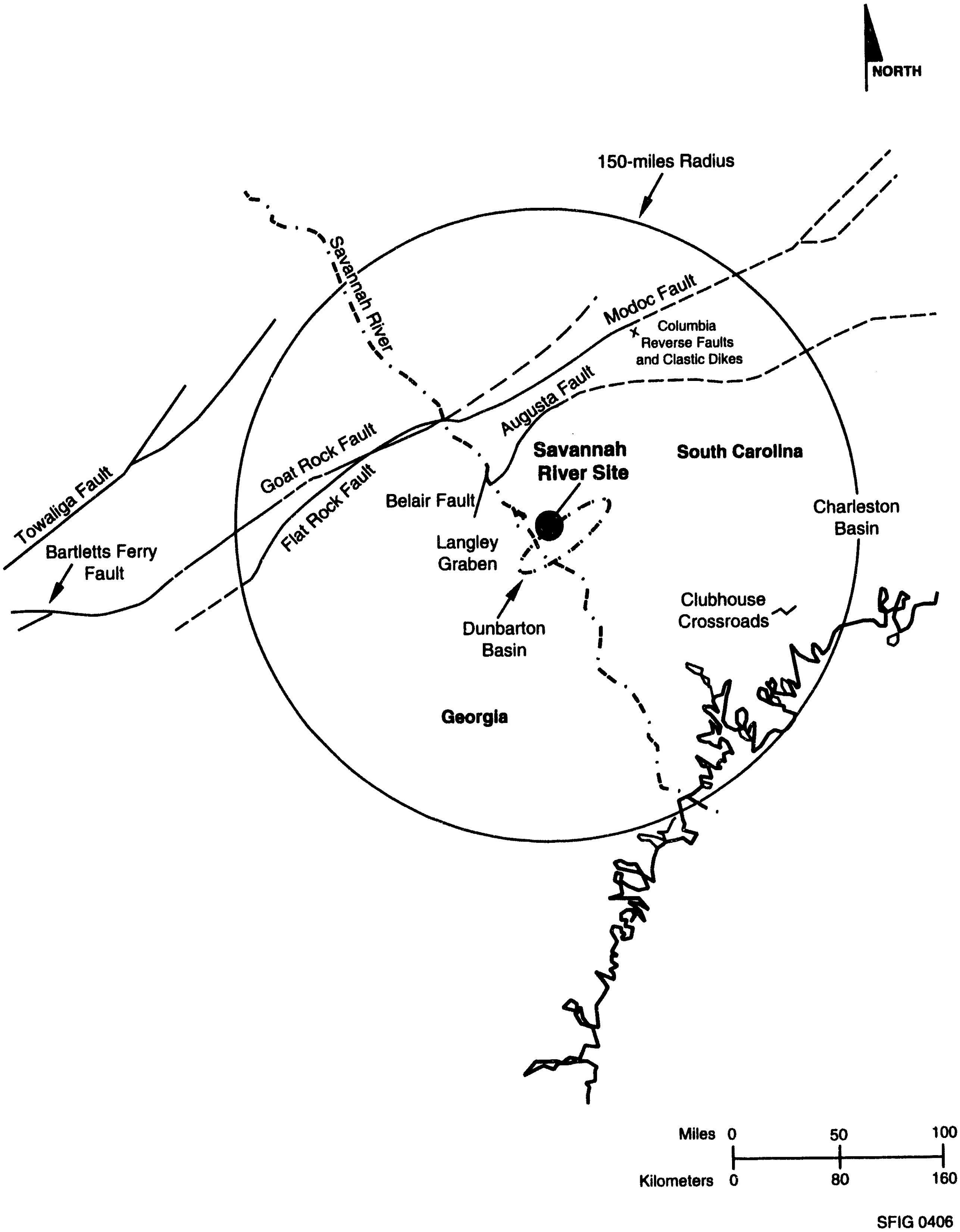


Figure 4-6. Geologic structures within 150 km of Savannah River Site.

Researchers have mapped the Pen Branch Fault for at least 24 kilometers (15 miles) across the central portion of the SRS (Snipes et al. 1993). This fault is probably a continuation of the northern boundary fault of the Triassic age Dunbarton basin and is interpreted as being at least a Cretaceous/Tertiary (144-1.6 million years) reactivation of that fault (WSRC 1993b). Observed displacements of the Coastal Plain sediments range from about 26 meters (85 feet) at the Basement/Cretaceous contact to about 9 meters (30 feet) in the shallower sediments (WSRC 1993b). Based on the available data, there is no evidence to indicate that the Pen Branch is a "capable fault" as defined by the U.S. Nuclear Regulatory Commission (NRC). Under the NRC definition, a fault is capable if it has moved within the last 35,000 years, has had recurring movement within the last 500,000 years, is related to any earthquake activity, or is associated with another capable fault. A recent study (Snipes et al. 1993) examined a Quaternary light tan soil horizon in SRS railroad cuts. The soil horizon, which has a thickness of 3 to 6 meters (10 to 20 feet), revealed no detectable offset, indicating that there has been no recent Pen Branch Fault activity. Figure 4-7 shows the locations of the Pen Branch Fault and other known or suspected faults within the Paleozoic and Triassic Basement (DOE 1991b).

Seismicity in the Coastal Plain of South Carolina occurs in three distinct seismic zones near the Charleston area (WSRC 1993b): Middleton Place-Summerville, about 19 kilometers (12 miles) northwest of Charleston; Bowman, about 59 kilometers (37 miles) northwest of the Middleton Place-Summerville; and Adams Run, about 30 kilometers (19 miles) southwest of the Middleton Place-Summerville (WSRC 1993b). Of the distinct seismic zones within the Coastal Plain province, the Charleston area has been and remains the most seismically active. The Charleston area is also the most significant source of seismicity affecting the SRS, both in terms of maximum historic site intensity and the number of earthquakes felt in the area (WSRC 1993b).

Tables 4-2 and 4-3 summarize the historic information on earthquakes that have occurred in the SRS region. Two notable earthquakes have occurred within 320 kilometers (200 miles) of the SRS. The first was a major earthquake in 1886 centered in the Charleston area about 145 kilometers (90 miles) from the Site; it had an estimated Richter magnitude of 6.8. DOE estimates that the SRS would have felt a tremor with an estimated Modified Mercalli Intensity (MMI) of VI to VII and an estimated peak horizontal acceleration of 10 percent of gravity, or 0.10*g*, due to that earthquake (WSRC 1993b). The second earthquake was the Union County, South Carolina, earthquake of 1913, which had an estimated Richter magnitude of 6.0 and occurred about 160 kilometers (100 miles) from the SRS (WSRC 1993b). This earthquake, which is the closest significant event to the SRS other than

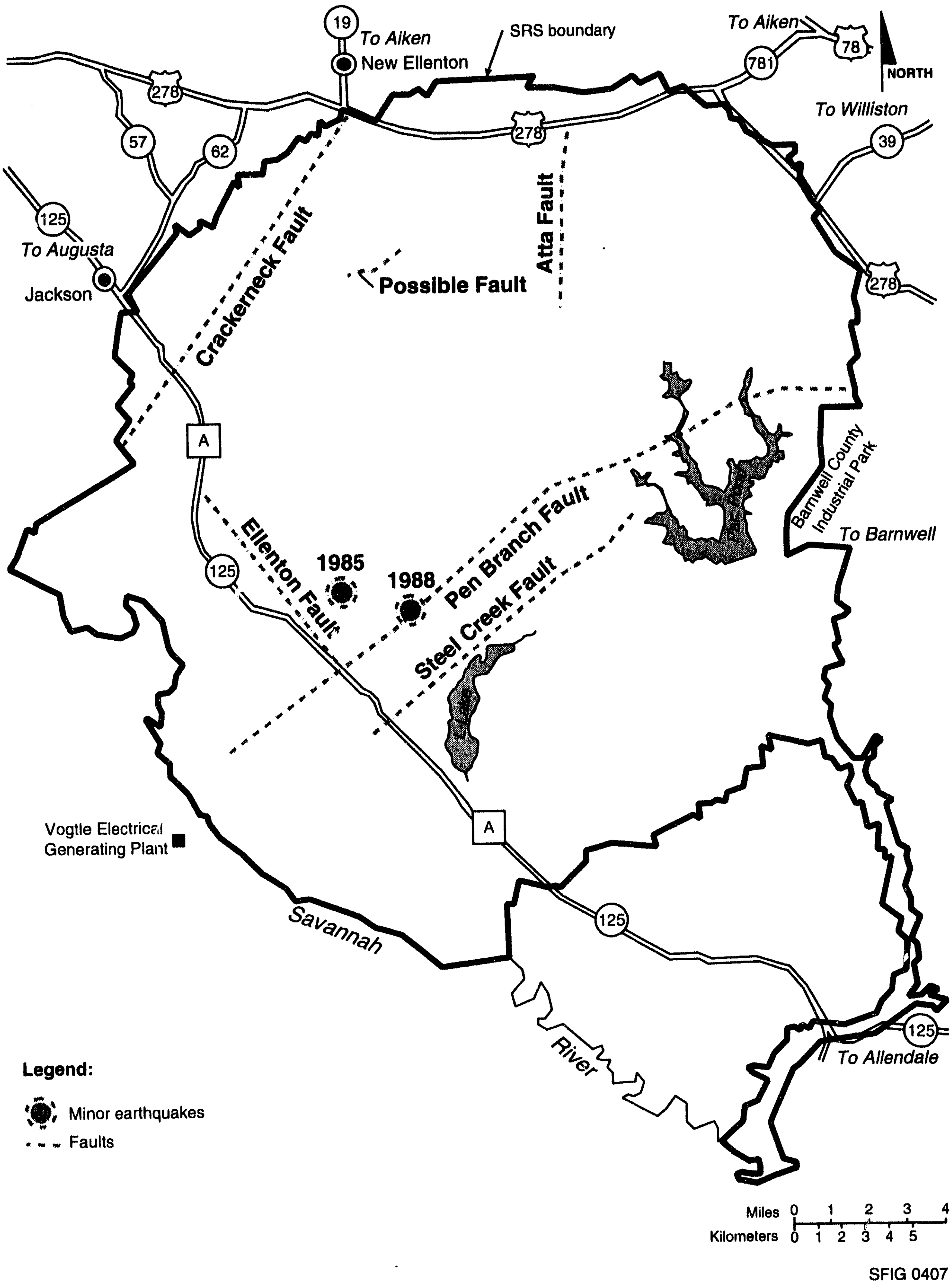


Figure 4-7. Geologic faults of the Savannah River Site.

Table 4-2. Earthquakes in the SRS region with a Modified Mercalli Intensity greater than V.[a]

Date[b]	Location	Coordinates Lat. (°N)	Coordinates Long. (°W)	Maximum Intensity	Distance from SRS (km)[c]	Reported or Estimated Intensity at SRS	Richter Magnitude	Estimated Acceleration at SRS(g)
1811 Jan 13	Burke Co., Ga.	33.2	82.2	V	55	III-IV	NA[d]	0.02
1811-1812 (3 shocks)	New Madrid, Mo.	36.3	89.5	XI-XII	850	V-VI	NA	0.05
1875 Nov 02	Lincolnton, Ga.	33.8	82.5	VI	100	III-IV	NA	0.02
1886 Sep 02	Charleston, S.C.	32.9	80.0	X	145	VI	6.8	0.10
1886 Oct 22	Charleston, S.C.	32.9	80.0	VII	155	III-IV	NA	0.02
1897 May 31	Giles Co., Va.	33.0	80.7	VIII	455	III	NA	0.02
1913 Jan 01	Union Co., S.C.	34.7	81.7	VII-VIII	160	IV	6.0[e]	0.02
1920 Aug 01	Charleston, S.C.	33.1	80.2	VII	135	III-IV	NA	0.02
1972 Feb 03	Bowman, S.C.	33.5	80.4	V	115	IV	4.5	0.02
1974 Aug 02	Willington, S.C.	33.9	82.5	VI	105	IV	4.1	0.02
1974 Nov 22	Charleston, S.C.	32.9	80.1	VI	145	III-IV	4.3	0.02

a. Source: DOE (1991b).
b. Based on Greenwich mean time.
c. Conversion factor: 1 kilometer = 0.6214 mile.
d. NA = data not available.
e. Estimated.

Table 4-3. Earthquakes in the SRS region with a Modified Mercalli Intensity greater than IV or a magnitude greater than 2.0.[a]

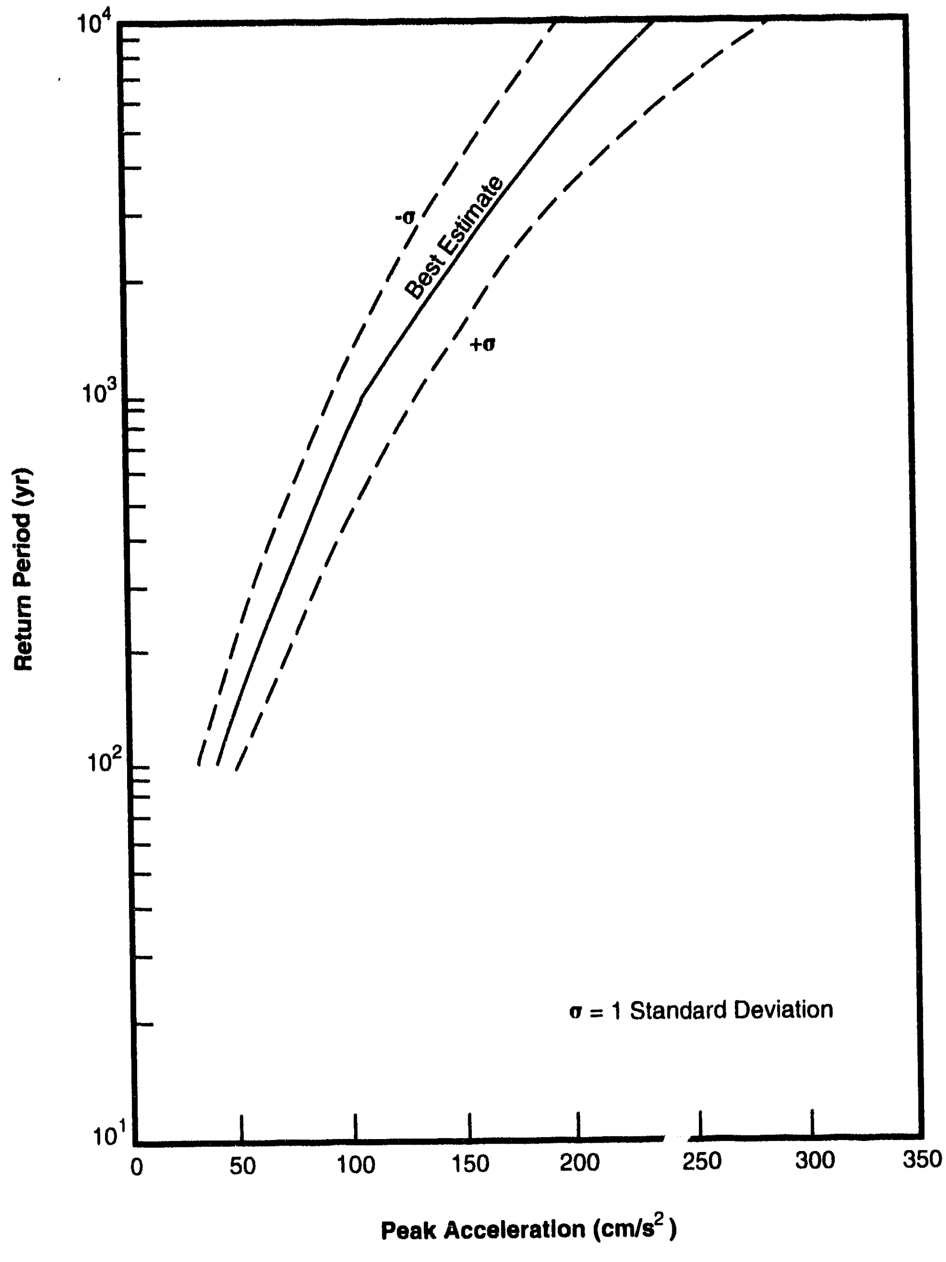
Date[b]	Coordinates Lat. (°N)	Long. (°W)	Maximum Intensity	Distance from SRS (km)[c]	Reported or Estimated Intensity at SRS	Richter Magnitude	Estimated Acceleration at SRS(*g*)
1811 Jan 13[d]	33.2	82.2	V	55	III-IV	NA[e]	0.02
1853 May 20	34.0	81.2	VI	102	NA	NA	NA
1945 Jul 26	33.8	81.4	V	77	NA	4.4	NA
1964 Mar 07	33.7	82.4	NA	85	NA	3.3	NA
1964 Apr 20	33.8	81.1	V	96	NA	3.5	NA
1968 Sep 22	34.1	81.5	IV	102	NA	3.5	NA
1972 Aug 14	33.2	81.4	NA	27	NA	3.0	NA
1974 Oct 28	33.8	81.9	IV	72	NA	3.0	NA
1974 Nov 05	33.7	82.2	III	77	NA	3.7	NA
1976 Sep 15	33.1	81.4	NA	25	NA	2.5	NA
1977 Jun 05	3.1	81.4	NA	35	NA	2.7	NA
1982 Jan 28	32.9	81.4	NA	40	NA	3.4	NA
1985 Jun 08	33.2	81.7	III	Onsite	III	2.6	NA
1988 Feb 17[f]	33.6	81.7	III	45	NA	2.6	NA
1988 Aug 05	33.1	81.4	NA	Onsite	II	2.0	NA
1993 Aug 08	TBD	TBD	TBD	TBD	TBD	3.2	NA

a. Source: DOE (1991b).
b. Based on Greenwich mean time.
c. Conversion factor: 1 kilometer = 0.6214 mile.
d. Located in Burke County, Ga.
e. NA = data not available.
f. Located at Aiken, S.C.

the Charleston-area earthquake, produced an estimated intensity of II to III (MMI) in the City of Aiken, which is approximately 19 kilometers (12 miles) north of the Site (DOE 1991b; WSRC 1993b).

Two earthquakes have occurred on the SRS during recent years (see Figure 4-7). On June 8, 1985, onsite instruments recorded an earthquake with a Richter magnitude of 2.6 and a focal depth of about 1.0 kilometer (0.6 mile) (WSRC 1993b). The epicenter was just west of the C- and K-Areas. The ground acceleration from this event did not activate instrumentation in the reactor areas (detection limits of 0.002*g*). On August 5, 1988, an earthquake with a Richter magnitude of 2.0 and a focal depth of approximately 2.7 kilometers (1.7 miles) occurred (Stephenson 1988); earthquakes of Richter magnitude 2.0 are normally detected only by specialized instrumentation. The epicenter for this event was just northeast of K-Area. Although this event was not felt by workers on the SRS, it was recorded by sensors within 96 kilometers (60 miles) of the Site. A report on the August 1988 earthquake (Stephenson 1988) also reviewed the latest earthquake history for the region. This report predicts recurrence period of 1 year for a magnitude 2.0 event for the southeast Coastal Plain. However, the report notes that historic data to calculate recurrence rates accurately are sparse. SRS workers did feel the effects of two other events that occurred in the area within the past 7 years. A Richter magnitude 2.6 earthquake occurred in the City of Aiken, approximately 19 kilometers (12 miles) north of the SRS on February 17, 1988. Reports indicate that this event was felt in the Aiken area and on the SRS (DOE 1991b). Most recently, a Richter magnitude 3.2 earthquake occurred on August 8, 1993, approximately 16 kilometers (10 miles) east of the City of Aiken near Couchton, South Carolina. Residents reported feeling this earthquake in Aiken, New Ellenton (immediately north of the SRS), North Augusta (approximately 40 kilometers [25 miles] northwest of the SRS), and the Site.

Based on seismic activity information in the past 300 years, this analysis does not project earthquakes greater than a Richter magnitude 6.0, which corresponds to a Modified Mercalli Intensity of VII, to occur on the SRS. The design-basis earthquake for the SRS is a Modified Mercalli Intensity VIII event, which corresponds to a horizontal peak ground acceleration of 0.2*g*. Based on current technology, as applied in various probabilistic evaluations of the seismic hazard in the SRS region, the 0.2*g* peak ground acceleration can be associated with a 2×10^{-4} annual probability of exceedance (5,000-year return period). This approach is consistent with the methodology accepted at commercial nuclear reactors (WSRC 1993b). Figure 4-8 shows seismic hazard curves for the SRS.



SFIG 0408

Figure 4-8. Seismic hazard curve for SRS.

A number of paleoliquefaction sites have been identified in Beaufort County, South Carolina, some 50 miles (80 kilometers) southeast of the SRS, indicating a likelihood of prehistoric seismic events outside of the currently-active Charleston seismic zone (Rajendran and Talwani 1993). There is no evidence to suggest that seismically-induced liquefaction of soils represents a hazard at SRS, however. Weak subsurface zones are encountered occasionally during drilling. These zones are associated with carbonate materials and appear to be related to dissolution of these materials.

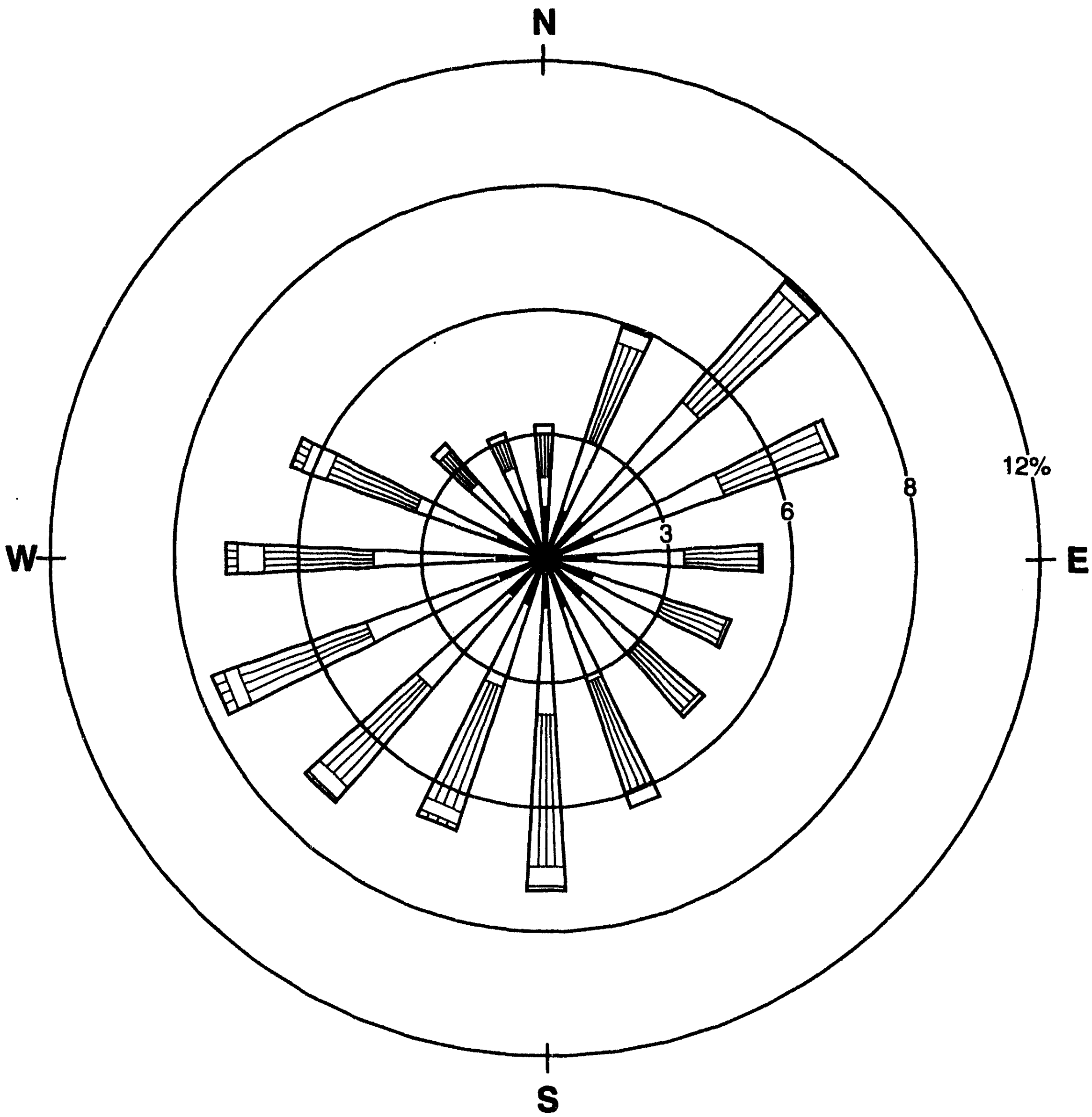
Engineering investigations have been conducted on granular soils underlying the Defense Waste Processing Facility [in S-Area just north of H-Area (see Figure 2-3)] to evaluate the cyclic mobility (liquefaction under cyclic stresses) of these soils (WSRC 1992b). These investigations determined that the sands and clayey sands throughout the subgrade will not experience liquefaction (strength loss leading to bearing capacity failures) and will not develop cyclic mobility (significant cyclic or accumulate deformations) under the safe shutdown earthquake with a peak horizontal ground surface acceleration of 0.20 *g* (9.8 meters/second^2 or 32.1 feet/second^2).

4.7 Air Resources

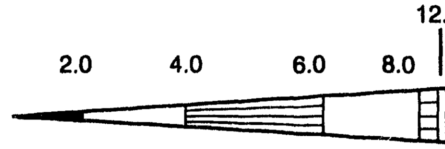
4.7.1 Meteorology and Climatology

The SRS collects wind data from instruments mounted on seven onsite 61-meter (200-foot) meteorological towers. Figure 4-9 shows a wind rose that represents annual wind direction frequencies and wind speeds for the SRS from 1987 through 1991. The maximum wind directional frequencies are from the northeast and west-southwest. The average wind speed for this 5-year period was 3.8 meters per second (8.5 miles per hour). Calm winds (less than 1 meter per second or 2.2 miles per hour) occurred less than 10 percent of the time during the 5-year period. Seasonally, wind speeds were greatest during the winter at 4.1 meters per second (9.5 miles per hour) and lowest during the summer at 3.4 meters per second (7.6 miles per hour) (Shedrow 1993).

The annual average temperature at the SRS is 18 degrees C (64 degrees F); monthly averages range from a low of 7 degrees C (45 degrees F) in January to a high of 27 degrees C (81 degrees F) in July. Relative humidity readings taken four times each day range from 36 percent in April to 98 percent in August (DOE 1991a).



The wind rose plot shows percent occurrence frequencies of wind direction and speed at SRS. It is based on a composite of hourly averaged wind data from the SRS meteorological tower network for the 5-year period 1987-1991. Measurements were taken from 200 feet above ground. Directions indicated are *from* which the wind blows.



Wind Speed Class Boundaries
(meters/second)

SFIG 0409

Figure 4-9. Wind rose for the Savannah River Site (1987-1991).

The average annual precipitation at the SRS is approximately 122 centimeters (48 inches). Precipitation distribution is fairly even throughout the year, with the highest precipitation in the summer [36.1 centimeters (14.2 inches)] and the lowest in autumn [22.4 centimeters (8.8 inches)]. Snowfall has occurred in the months of October through March, with the average annual snowfall at 3.0 centimeters (1.2 inches). Large snowfalls are rare (DOE 1991a).

Winter storms in the SRS area occasionally bring strong and gusty surface winds with speeds as high as 32 meters per second (72 miles per hour). Thunderstorms can generate winds with speeds as high as 18 meters per second (40 miles per hour) and even stronger gusts. The fastest 1-minute wind speed recorded at Augusta between 1950 and 1986 was 37 meters per second (83 miles per hour (DOE 1991a).

4.7.1.1 Occurrence of Violent Weather. The SRS area experiences an average of 56 thunderstorm days per year. From 1954 to 1983, 37 tornadoes were reported for a 1-degree square of latitude and longitude that includes the SRS (DOE 1991a). This frequency of occurrence is equivalent to an average of about one tornado per year. The estimated probability of a tornado striking a point on the SRS is 7×10^{-5} per year (DOE 1991a). Since operations began at the SRS in 1953, nine confirmed tornadoes have occurred on or near the Site. They caused nothing more than light damage, with the exception of a tornado in October 1989 that caused considerable damage to forest resources in an undeveloped southeastern sector of the SRS (Shedrow 1993).

From 1700 to 1992, 36 hurricanes occurred in South Carolina, resulting in an average frequency of about one hurricane every 8 years. Three hurricanes were classified as major. Because SRS is about 160 kilometers (100 miles) inland, the winds associated with hurricanes have usually diminished below hurricane force [i.e., equal to or greater than a sustained wind speed of 33.5 meters per second (75 miles per hour)] before reaching the SRS. Winds exceeding hurricane force have been observed only once at SRS (Hurricane Gracie in 1959) (Shedrow 1993).

4.7.1.2 Atmospheric Stability. Based on measurements at onsite meteorological stations, the atmosphere in the SRS region is unstable approximately 56 percent of the time, neutral 23 percent of the time, and stable about 21 percent of the time. On an annual basis, inversion conditions occur 21 percent of the time at the SRS (Shedrow 1993).

4.7.2 Nonradiological Air Quality

4.7.2.1 Background Air Quality. The SRS is in the Augusta (Georgia) - Aiken (South Carolina) Interstate Air Quality Control Region (AQCR). This Air Quality Control Region, which is designated as a Class II area, is in compliance with National Ambient Air Quality Standards (NAAQS) for criteria pollutants. The criteria pollutants include sulfur dioxide, nitrogen oxides reported as nitrogen dioxide, particulate matter (less than or equal to 10 microns), carbon monoxide, ozone, and lead (CFR 1993a). The closest nonattainment area to the SRS is the Atlanta, Georgia, air quality region, 233 kilometers (145 miles) to the west, which is in nonattainment of the standard for ozone.

The SRS will have to comply with Prevention of Significant Deterioration (PSD) Class II requirements if there is a significant increase in emissions of criteria air pollutants due to a modification at the Site (CFR 1993b). Development at the SRS has not yet triggered Prevention of Significant Deterioration permitting requirements. If a permit were required, the SRS would have to address several requirements, including impacts on the air quality of Class I areas within 10 kilometers (6.2 miles) of the Site [40 CFR Part 52.21(b)(23)(iii)]. The nearest Class I area to the SRS is the Congaree Swamp National Monument in South Carolina, approximately 73 kilometers (45 miles) to the east-northeast of the Site. Therefore, a Prevention of Significant Deterioration permit, if required for the SRS, would not have to address Class I areas.

4.7.2.2 Air Pollutant Source Emissions. The SRS utilized the 1990 comprehensive emissions inventory data to establish the baseline year for showing compliance with State and Federal air quality standards - calculating both maximum potential and actual emission rates. The air quality compliance demonstration also included sources forecast for construction or operation in this decade (for which the SRS had obtained air quality construction permits through December 1992). The SRS based its calculated emission rates for the sources on process knowledge, source testing, permitted operating capacity, material balance, and U.S. Environmental Protection Agency (EPA) Air Pollution Emission Factors (AP-42; EPA 1985).

4.7.2.3 Ambient Air Monitoring. At present, the SRS performs no onsite ambient air quality monitoring. State agencies operate ambient air quality monitoring sites in Barnwell, Aiken, and Richmond Counties. These areas, which include the SRS, are in attainment with National Ambient Air Quality Standards for sulfur dioxide, nitrogen oxides, carbon monoxide, particulate matter, ozone, and lead (CFR 1993a).

4.7.2.4 Atmospheric Dispersion Modeling. The SRS has performed atmospheric dispersion modeling for criteria and air toxic air pollutants for both maximum potential and actual emissions for the base year 1990, using the EPA Industrial Source Complex Short Term No. 2 Model. The SRS used 1991 meteorological data collected at the Site meteorological stations for input to the model.

4.7.2.5 Summary of Nonradiological Air Quality. The SRS is in compliance with National Ambient Air Quality Standards and with the gaseous fluoride and total suspended particulate standards required by South Carolina Department of Health and Environmental Control (SCDHEC) Regulation R.61-62.5, Standard 2, "Ambient Air Quality Standards" (AAQS) (see Table 4-4).

The South Carolina Department of Health and Environmental Control has non-radiological air quality regulatory authority over the SRS. The Department determines SRS ambient air quality compliance based on SRS air pollutant emissions modeled at the Site perimeter (excluding SC Highway 125, which crosses the southwestern quadrant of the SRS).

The SRS is in compliance with Department of Health and Environmental Control Regulation R.61-62.5, Standard 8, "Toxic Air Pollutants," which regulates the emission of 257 toxic substances. The SRS has identified emission sources for 139 of the 257 regulated substances; the modeled results indicate that the Site is within applicable Department of Health and Environmental Control standards (WSRC 1993c). Table 4-5 lists SRS emissions of toxic air pollutants of concern related to the SRS spent nuclear fuel alternatives, based on 1990 baseline data and the potential sources of air pollution permitted for construction or operation in December 1992.

4.7.3 Radiological Air Quality

4.7.3.1 Background and Baseline Radiological Conditions. In the SRS region, airborne radionuclides originate from natural resources (terrestrial or cosmic), worldwide fallout, and Site operations. The SRS maintains a network of air monitoring stations on and around the Site to determine concentrations of radioactive particulates and aerosols in the air (Arnett et al. 1992). Table 4-6 lists average and maximum atmospheric radionuclide concentrations at the SRS boundary and background [160-kilometer (100-mile) radius] monitoring locations during 1991. Table 4-7 lists the average concentrations of tritium in the atmosphere, as measured at on- and offsite monitoring locations.

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Table 4-4. Estimated ambient concentration contributions of criteria air pollutants from existing SRS sources and sources planned for construction or operation through 1995 ($\mu g/m^3$).[a,b]

Pollutant[c]	Averaging time	SRS Maximum Potential Concentration	Actual	Most stringent AAQS[d] (Federal or state)	Maximum Potential Concentration as a Percent of AAQS[e]
SO_2	Annual	18	10	80[f]	22.5
	24-hour	356	185	365[f,g]	97.5
	3-hour	1,210	634	1,300[f,g]	93
NO_x	Annual	30	4	100[f]	30
CO	8-hour	818	23	10,000[f,g]	8
	1-hour	3,553	180	40,000[f,g]	9
Gaseous fluorides (as HF)	12-hour	2.40	0.62	3.7[e]	65
	24-hour	1.20	0.31	2.9[e]	41
	1-week	0.6	0.15	1.6[e]	38
	1-month	0.11	0.03	0.8[e]	14
PM_{10}	Annual	9	3	50[f]	18
	24-hour	93	56	150[f]	62
O_3	1-hour	NA	NA	235[f,g]	NA
TSP	Annual geometric mean	20	11	75[e]	2.7
Lead	Calendar quarter mean	0.0015	0.0003	1.5[e]	0.1

a. Source: WSRC (1994a).
b. The contributions listed are the maximum values at the SRS boundary.
c. SO_2 = sulfur dioxide; NO_x = nitrogen oxides; CO = carbon monoxide; PM_{10} = particulate matter $\leq$ 10μm in diameter; TSP = Total Suspended Particulates, O_3 = Ozone.
d. AAQS = Ambient Air Quality Standard.
e. Source: South Carolina Department of Health and Environmental Control, R.61-62.
f. Source: 40 CFR Part 50.
g. Concentration not to be exceeded more than once a year.
NA = Not available.

Table 4-5. Baseline 24-hour average modeled concentrations at the SRS boundary - toxic air pollutants regulated by South Carolina from existing SRS sources and sources planned for construction or operation through 1995 ($\mu g/m^3$).[a]

Pollutant[b]	Regulatory Limit	Maximum Potential Concentration[c]	Actual Concentration[d]	Maximum Potential Concentration as a Percent of AAQS[e]
Nitric acid	125	51	4.0	41
1,1,1-Trichloroethane	9,550	81	22	1
Benzene	150	32	31	21
Ethanolamine	200	<0.01	<0.01	<0.1
Ethyl benzene	4,350	0.58	0.12	<0.1
Ethylene glycol	650	0.20	0.08	<0.1
Formaldehyde	7.5	<0.01	<0.01	<0.1
Glycol ethers	Pending	<0.01	<0.01	—
Hexachloronapthalene	1	<0.01	<0.01	<0.1
Hexane	200	0.21	0.072	<0.1
Manganese	25	0.82	0.10	3
Methyl alcohol	1,310	2.9	0.51	0.2
Methyl ethyl ketone	14,750	6.0	0.99	<0.1
Methyl isobutyl ketone	2,050	3.0	0.51	<0.1
Methylene chloride	8,750	10.5	1.8	<0.1
Naphthalene	1,250	0.01	0.01	<0.1
Phenol	190	0.03	0.03	<0.1
Phosphorus	0.5	<0.001	<0.001	<0.1
Sodium hydroxide	20	0.01	0.01	<0.1
Toluene	2,000	9.3	1.6	<0.1
Trichloroethylene	6,750	4.8	1.0	<0.1
Vinyl acetate	176	0.06	0.02	<0.1
Xylene	4,350	39	3.8	0.9

a. Source: WSRC (1994a).
b. Pollutants listed include compounds of interest regarding spent nuclear fuel alternatives.
c. Maximum potential emissions from all SRS sources for 1990 plus maximum potential emissions for sources permitted in 1991 and 1992.
d. Actual emissions from all SRS sources plus maximum potential emissions for sources permitted for construction through December 1992.
e. AAQS = Ambient Air Quality Standard.

Table 4-6. Radioactivity in air at SRS perimeter and at 160-kilometer (100-mile) radius (pCi/m^3).[a]

Location	Gross Alpha	Nonvolatile Beta	Sr-89,90[b]	Pu-238[b]	Pu-239[b]
Site perimeter					
Average	2.61×10^{-3}	1.78×10^{-2}	4.90×10^{-5}	1.22×10^{-6}	2.11×10^{-6}
Maximum	1.07×10^{-2}	4.63×10^{-2}	5.11×10^{-4}	1.94×10^{-5}	5.40×10^{-5}
Background (160-kilometer radius)					
Average	2.60×10^{-3}	1.76×10^{-2}	2.00×10^{-4}	1.44×10^{-6}	6.10×10^{-7}
Maximum	9.31×10^{-3}	5.26×10^{-2}	2.08×10^{-3}	2.39×10^{-5}	5.40×10^{-6}

a. Source: Arnett et al. (1992).
b. Monthly composite.

Table 4-7. Average atmospheric tritium concentrations on and around the Savannah River Site (pCi/m^3).[a]

Location	1991	1990	1989
Onsite	250	430	640
Site perimeter	21	32	37
40-kilometer radius	11	12	14
160-kilometer radius	8.5	8.8	9

a. Source: Arnett et al. (1992).

4.7.3.2 Sources of Radiological Emissions. Table 4-8 lists groups of facilities that released radionuclides to the atmosphere in 1992; the facilities are grouped according to the principal function that resulted in the release of radioactive materials.

Table 4-9 lists both the identified radionuclides that contributed to the SRS dose and the percent contribution of each radionuclide to the total site effective dose equivalent.

Table 4-8. Operational groupings and function of radionuclide sources.

Group	Function
Reactor Materials	Production of fuel and targets
Reactors	Irradiation of fuel and targets
Separations	Separation of useful radionuclides (other than tritium)
Analytical Laboratories	Process Control Laboratories
Tritium	Extraction, purification, and packaging
Waste Management	Management of radioactive waste
Savannah River Technology Center	Research and development to support SRS processes

4.8 Water Resources

4.8.1 Surface Water

The Savannah River bounds the SRS on its southwestern border for about 20 miles (32 kilometers), approximately 160 river miles (260 kilometers) from the Atlantic Ocean. At the SRS, river flow averages about 10,000 cubic fect (283 cubic meters) per second. River flows range from 3,960 cubic feet (112 cubic meters) per second to 71,700 cubic feet (2,030 cubic meters).

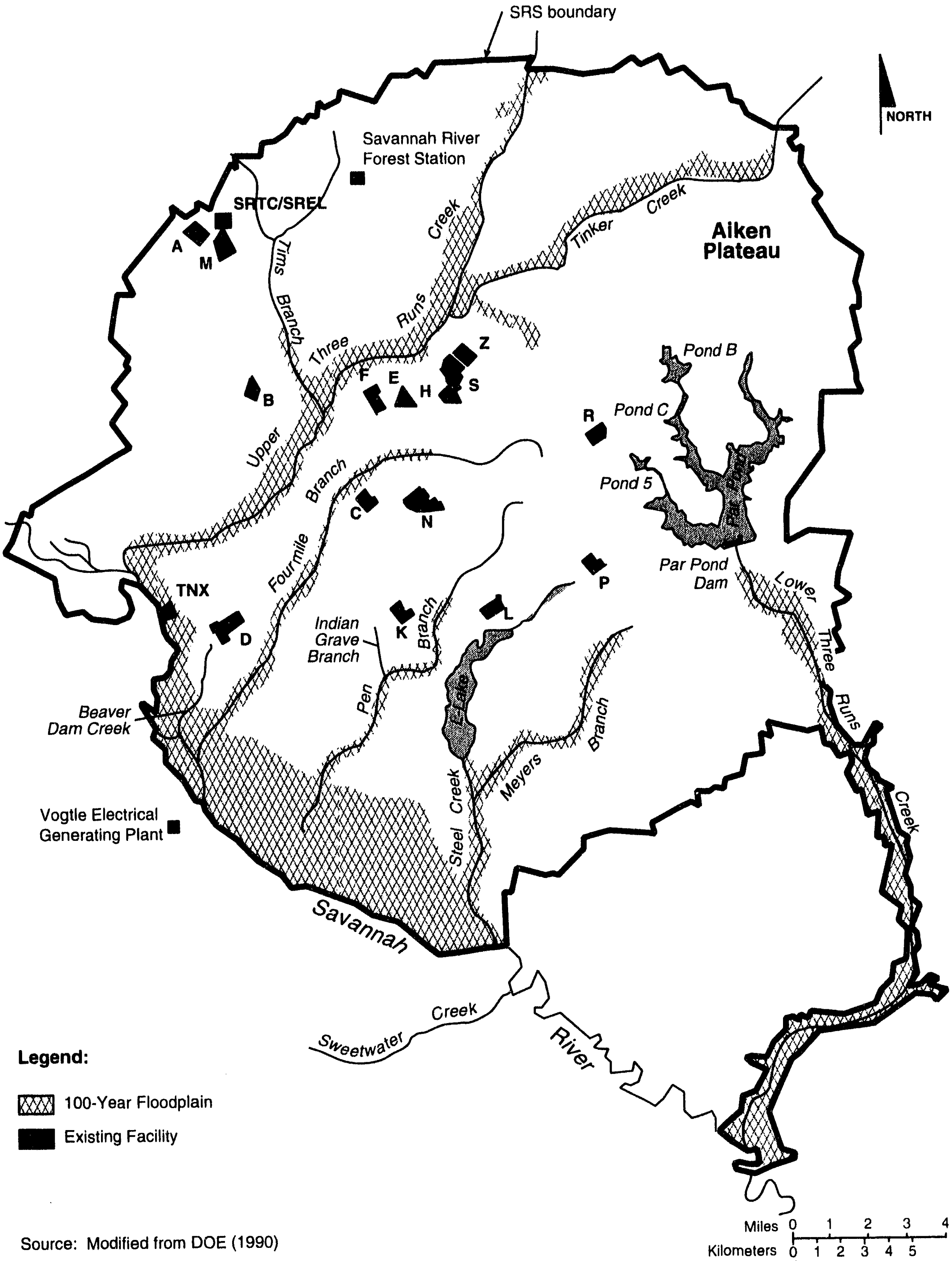
Five upstream reservoirs - Jocassee, Keowee, Hartwell, Richard B. Russell, and Strom Thurmond - minimize the effects of droughts and the impacts of low flow on downstream water quality and fish and wildlife resources in the river.

At the SRS, a swamp occupies the floodplain along the Savannah River for a distance of approximately 10 miles (17 kilometers); the swamp is about 1.5 miles (2.5 kilometers) wide. A natural levee separates the river from the swampy floodplain. Figure 4-10 shows the 100-year floodplain of the Savannah River in the vicinity of the SRS as well as the floodplains of major tributaries draining the SRS. A 500-year floodplain map of the SRS has not been completed, but would be required prior to the siting of any spent nuclear fuel management facilities, in compliance with DOE regulations (10 CFR Part 1022, "Compliance with Floodplain/Wetlands Environmental Review Requirements"). These regulations require DOE to evaluate the potential effects of flooding to

Table 4-9. Annual quantity of radionuclide emissions from the Savannah River Site.[a,b]

Radionuclide	Annual Quantity (curies)	Percent of Total Site Dose
H-3 (oxide)	1.00×10^{5}	98.0
Pu-239	7.45×10^{-4}	0.6
U-235,238	1.58×10^{-3}	0.4
Pu-238	4.46×10^{-4}	0.3
Ar-41	2.51×10^{2}	0.3
I-129	3.50×10^{-3}	0.2
Am-241,243	1.13×10^{-4}	0.1
Sr-89,90 (Y-90)	2.03×10^{-3}	0.02
Cm-242,244	2.31×10^{-5}	0.01
Cs-137 (Ba-137m)	2.50×10^{-4}	0.01
C-14	1.86×10^{-1}	0.01
H-3 (elemental)	5.59×10^{4}	<0.01
I-135	1.34×10^{-1}	<0.01
Kr-85	4.99×10^{1}	<0.01
I-131	9.99×10^{-5}	<0.01
Ru-106 (Rh-106)	1.81×10^{-6}	<0.01
I-133	1.15×10^{-3}	<0.01
Co-60	3.60×10^{-7}	<0.01
Xe-135	2.43×10^{-3}	<0.01
Cs-134	3.75×10^{-8}	<0.01
Ce-144 (Pr-144,144m)	1.16×10^{-7}	<0.01
Eu-154	3.44×10^{-13}	<0.01
Eu-155	1.63×10^{-13}	<0.01
Sb-125	7.27×10^{-15}	<0.01
Zr-95 (Nb-95)	2.39×10^{-14}	<0.01

a. Source: Arnett et al. (1993).
b. Includes emissions to the atmosphere and surface water.



Source: Modified from DOE (1990)

Figure 4-10. Savannah River Site, showing 100-year floodplain, major stream systems and facilities.

proposed "critical actions" (for example, the storage of highly toxic or water-reactive materials), which it defines as those for which even a slight chance of flooding would be unacceptable.

The five principal tributaries to the river on the SRS are Upper Three Runs Creek, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs Creek (Figure 4-10). These tributaries drain almost all of the SRS. Each of these streams originates on the Aiken Plateau in the Coastal Plain and descends 50 to 200 feet (15 to 60 meters) before discharging into the river. The streams, which historically have received varying amounts of effluent from various SRS operations, are not commercial sources of water. The natural flow of SRS streams ranges from less than 10 cubic feet (1 cubic meter) per second in smaller streams such as Pen Branch to 240 cubic feet (6.8 cubic meters) per second in Upper Three Runs Creek.

4.8.1.1 SRS Streams. This section describes the pertinent physical and hydrologic properties of Upper Three Runs Creek and Fourmile Branch, which are the streams closest to most SRS spent nuclear fuel management locations (Figure 4-10). These two streams are among the largest on the SRS, and they border the areas where DOE is most likely to locate new spent nuclear fuel facilities.

Upper Three Runs Creek is a large, cool [annual maximum temperature of 26.1 degrees C (79 degrees F)] blackwater stream in the northern part of the SRS. It drains an area of approximately 210 square miles (545 square kilometers), and has an average discharge of 330 cubic feet (9.3 cubic meters) per second at the mouth of the creek. Upper Three Runs Creek is approximately 25 miles (40 kilometers) long, with its lower 17 miles (28 kilometers) inside the boundaries of the SRS. This creek receives more water from underground sources than the other SRS streams and, therefore, has low conductivity, hardness, and pH values. Upper Three Runs Creek is the only major tributary on the SRS that has never received thermal discharges.

Fourmile Branch is about 15 miles (24 kilometers) long and drains an area of approximately 34 square miles (89 square kilometers). In its headwaters, Fourmile Branch is a small blackwater stream that receives relatively few impacts from SRS operations. The water chemistry in the headwater area of the creek is very similar to that of Upper Three Runs Creek, with the exception of nitrate concentrations, which are an order of magnitude higher than those in Upper Three Runs Creek (WSRC 1993e). These elevated nitrate concentrations are probably the result of groundwater transport and outcropping from the F- and H-Area seepage basins. In its lower reaches, Fourmile Branch broadens and flows through a delta formed by the deposition of sediments. Although most of the flow through the delta is in one main channel, the delta has many standing dead trees, logs, stumps, and cypress trees that provide structure and reduce the water velocity in some areas. Downstream of the delta, the creek flows in one main channel and most of the flow discharges into the Savannah River at River Mile 152 (kilometer 245), while a small portion of the creek flows west and enters Beaver Dam Creek, a small onsite tributary.

4.8.1.2 Surface-Water Quality. The Savannah River, which forms the boundary between the States of Georgia and South Carolina, supplies potable water to several users. Upstream of the SRS, the river supplies domestic and industrial water needs for Augusta, Georgia, and North Augusta, South Carolina. The river also receives sewage treatment plant effluent from Augusta, Georgia; North Augusta, Aiken, and Horse Creek Valley, South Carolina; and as described above from a variety of SRS operations via onsite stream discharges. Approximately 130 river-miles (210 kilometers) downstream of the SRS, the river supplies domestic and industrial water needs for Savannah, Georgia, and Beaufort and Jasper Counties in South Carolina through intakes located at about River Mile 29 and River Mile 39. In addition, Georgia Power's Vogtle Electric Generating Plant withdraws an average of 1.3 cubic meters per second (46 cubic feet per second) for cooling and returns an average of 0.35 cubic meters per second (12 cubic feet per second) of cooling tower blowdown. Also, the Urquhart Steam Generating Station at Beech Island, South Carolina withdraws approximately 7.5 cubic meters per second (265 cubic feet per second) of once through cooling water.

The South Carolina Department of Health and Environmental Control regulates the physical properties and concentrations of chemicals and metals in SRS effluents under the National Pollutant Discharge Elimination System (NPDES) program. This agency also regulates chemical and biological water quality standards for SRS waters. On April 24, 1992, the agency changed the classification of the Savannah River and SRS streams from "Class B waters" to "Freshwaters." The definitions of Class B waters and Freshwaters are the same, but the Freshwaters classification imposes a more stringent set of water quality standards (Arnett et al. 1993). Tables 4-10 and 4-11 list the characteristics of SRS surface-water quality upstream and downstream, respectively, due to contributions from SRS and possibly other sources. A comparison of these results indicates that influences from SRS or other sources are not seriously degrading Savannah River water quality.

Table 4-10. Water quality in the Savannah River above the confluence with Upper Three Runs near the Savannah River Site in 1990.[a,b]

Parameter	Unit of Measure	MCL [c,d] or DCG[e]	Existing Water-Body Concentration[f]	
			Average	Maximum
Aluminum	mg/L	0.05-0.2[g]	NC[i]	1.1
Ammonia	mg/L	NA[j]	0.1	0.2
Cadmium	mg/L	0.005[g]	NC	<0.01
Calcium	mg/L	NA	NC	4.4
Cesium-137	pCi/L	120[e]	0.0088	0.030
Chemical oxygen demand	mg/L	NA	9.7	17
Chloride	mg/L	250[h]	7.8	11
Chromium	mg/L	0.1[d]	NC	<0.02
Copper	mg/L	1.0[d]	NC	<0.01
Dissolved oxygen	mg/L	>5	8.0	9.6
Fecal coliform	Colonies per 100/ml	1,000[g]	54	197
Gross alpha	pCi/L	15g	0.04	0.36
Iron[c]	mg/L	0.3[h]	NC	1.5
Lead	mg/L	0.015[g]	NC	0.27
Magnesium	mg/L	NA	NC	1.4
Manganese[c]	mg/L	0.05[g]	NC	0.12
Mercury	mg/L	0.002[d]	NC	<0.0002
Nickel	mg/L	0.1[c]	NC	<0.05
Nitrite/Nitrate	mg/L	10[g]	0.32	0.99
Nonvolatile beta (dissolved)	pCi/L	50[g]	1.9	3.6
pH	pH Units	6.5-8.5[g]	Not reported	7.4
Phosphate	mg/L	N/A	0.09	0.16
Plutonium-238	pCi/L	1.6[e]	0.0006	0.0021
Plutonium-239	pCi/L	1.2[e]	0.0005	0.0021
Sodium	mg/L	NA	NC	11
Strontium-89	pCi/L	800[e]	0.23	1.0
Strontium-90	pCi/L	8[c]	0.09	0.22
Sulfate	mg/L	250[h]	7.8	11
Suspended solids	mg/L	NA	13	22
Temperature	Degrees Celsius	32.2[k]	18.0	27
Total dissolved solids	mg/L	500[h]	62	76
Tritium	pCi/L	20,000[c]	150	1,110
Zinc	mg/L	5[h]	NC	0.02

a. Source: Cummins et al. (1991).
b. Parameters are those for which DOE routinely measures as a regulatory requirement or as part of ongoing monitoring programs.
c. Maximum Contaminant Level (MCL), EPA National Primary Drinking Water Regulations (40 CFR Part 141).
d. Maximum Contaminant Level (MCL); South Carolina (1976).
e. U.S. Department of Energy Derived Concentration Guides (DCGs) for Water (DOE 1993c). DCG values are based on committed effective dose of 100 millirem per year; however, because drinking water MCL is based on 4 millirem per year, number listed is 4 percent of DCG.
f. Average concentration of samples taken at downstream monitoring station. Maximum is highest sampled concentration along reach of river potentially affected by site activities. Less than (<) indicates concentration below analysis detection limit.
g. Concentration exceeded water quality criteria; however, these criteria are listed for comparison only. Similarly, drinking water standards and DOE DCGs are listed. Water Quality Criteria (WQCs) and secondary standards are not legally enforceable.
h. Secondary Maximum Contaminant Level (SMCL), EPA National Secondary Drinking Water Regulations (40 CFR Part 143).
i. NC = Not calculated due to insufficient number of samples.
j. NA = None applicable.
k. Shall not exceed weekly average of 32.2 degrees Celsius after mixing nor rise more than 2.8 degrees Celsius in 1 week unless appropriate temperature criterion mixing zone has been established.

Table 4-11. Water quality in the Savannah River below the confluence with Lower Three Runs near the Savannah River Site in 1990.[a,b]

Parameter	Unit of Measure	MCL [c,d] or DCG[e]	Existing Water-Body Concentration[f]	
			Average	Maximum
Aluminum	mg/L	0.05-0.2[g]	NC[i]	1.1
Ammonia	mg/L	NA[j]	0.1	0.2
Cadmium	mg/L	0.005[g]	NC	<0.01
Calcium	mg/L	NA	NC	4.4
Cesium-137	pCi/L	120[e]	0.028	0.037
Chemical oxygen demand	mg/L	NA	9.8	14
Chloride	mg/L	250[h]	8	10
Chromium	mg/L	0.1[d]	NC	<0.02
Copper	mg/L	1.0[d]	NC	<0.01
Dissolved oxygen	mg/L	>5	7.7	9.5
Fecal coliform	Colonies per 100/ml	1,000[g]	54	197
Gross alpha	pCi/L	15g	0.08	1.48
Iron[c]	mg/L	0.3[h]	NC	1.5
Lead	mg/L	0.015[g]	NC	0.01
Magnesium	mg/L	NA	NC	1.3
Manganese[c]	mg/L	0.05[h]	NC	0.1
Mercury	mg/L	0.002[d]	NC	<0.0002
Nickel	mg/L	0.1[c]	NC	<0.05
Nitrite/Nitrate	mg/L	10[g]	0.28	0.43
Nonvolatile beta (dissolved)	pCi/L	50[g]	2.1	5.1
pH	pH Units	6.5-8.5[h]	Not reported	8.2
Phosphate	mg/L	N/A	0.1	0.16
Plutonium-238	pCi/L	1.6[e]	0.0006	0.0029
Plutonium-239	pCi/L	1.2[e]	0.0014	0.0079
Sodium	mg/L	NA	NC	11
Strontium-89	pCi/L	800[e]	0.25	0.98
Strontium-90	pCi/L	8[c]	0.13	0.30
Sulfate	mg/L	250[h]	8.5	12
Suspended solids	mg/L	NA	12	19
Temperature	Degrees Celsius	32.2[k]	18.0	27
Total dissolved solids	mg/L	500[h]	63	71
Tritium	pCi/L	20,000[c]	900	6,810
Zinc	mg/L	5[h]	NC	0.02

a. Source: Cummins et al. (1991).
b. Parameters are those for which DOE routinely measures as a regulatory requirement or as part of ongoing monitoring programs.
c. Maximum Contaminant Level (MCL), EPA National Primary Drinking Water Regulations (40 CFR Part 141).
d. Maximum Contaminant Level (MCL); South Carolina (1976).
e. U.S. Department of Energy Derived Concentration Guides (DCGs) for Water (DOE 1993c). DCG values are based on committed effective dose of 100 millirem per year; however, because drinking water MCL is based on 4 millirem per year, number listed is 4 percent of DCG.
f. Average concentration of samples taken at downstream monitoring station. Maximum is highest sampled concentration along reach of river potentially affected by site activities. Less than (<) indicates concentration below analysis detection limit.
g. Concentration exceeded water quality criteria; however, these criteria are listed for comparison only. Similarly, drinking water standards and DOE DCGs are listed. Water Quality Criteria (WQCs) and secondary standards are not legally enforceable.
h. Secondary Maximum Contaminant Level (SMCL), EPA National Secondary Drinking Water Regulations (40 CFR Part 143).
i. NC = Not calculated due to insufficient number of samples.
j. NA = None applicable.
k. Shall not exceed weekly average of 32.2 degrees Celsius after mixing nor rise more than 2.8 degrees Celsius in 1 week unless appropriate temperature criterion mixing zone has been established.

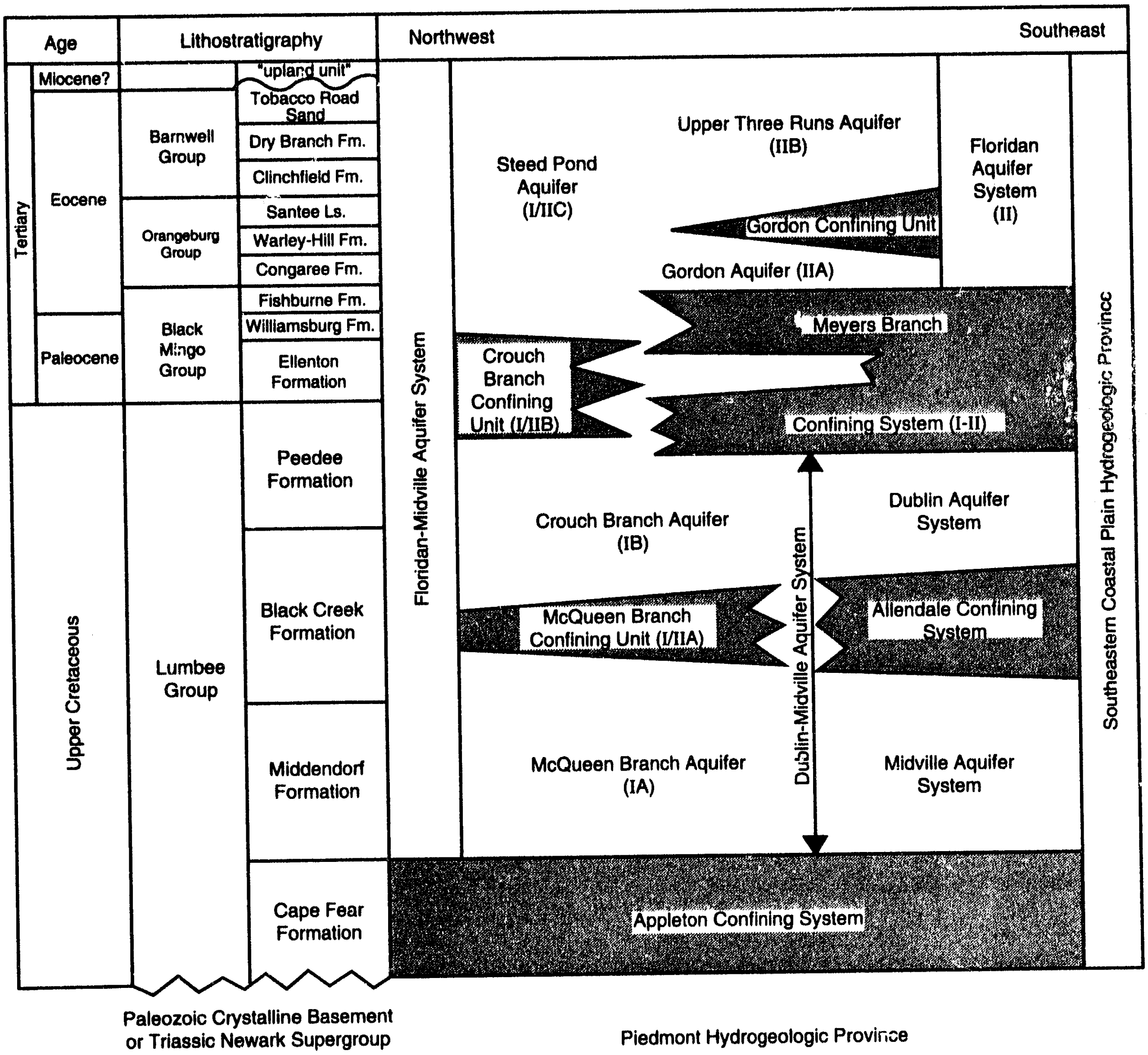
4.8.2 Groundwater Resources

4.8.2.1 Hydrostratigraphic Units. There are two hydrogeologic provinces in the subsurface beneath SRS (WSRC 1993e). The first, referred to as the Piedmont hydrogeologic province (Figure 4-11), includes Paleozoic metamorphic and igneous basement rocks and Triassic-aged lithified mudstone, sandstone, and conglomerate contained within the Dunbarton Basin. The second, referred to as the Southeastern Coastal Plain hydrogeologic province, represents the major aquifer systems and consists of a wedge of unconsolidated Coastal Plain sediments of Late Cretaceous and Tertiary age (Figure 4-11). These two units are overlain by the vadose or unsaturated zone, which extends from the ground surface to the water table. The unsaturated zone is a heterogeneous unit of clean, clayey, or silty sand through which recharge takes place.

The sediments that make up the Southeastern Coastal Plain hydrogeologic province in west-central South Carolina are grouped into three major aquifer systems divided by two major confining systems, all of which are underlain by the Appleton confining system (Figure 4-11). The Appleton system separates the Southeastern Coastal Plain hydrogeologic province from the underlying Piedmont hydrogeologic province. Locally, each of the major aquifer systems contains individual aquifer and confining units. Figure 4-11 shows the regional lithostratigraphy of the geologic province with the attendant primary hydrostratigraphic subdivision of the province. The complexly interbedded strata that form the three aquifer systems consist primarily of fine- to coarse-grained sand and local gravel and limestone deposited under relatively high energy conditions in fluvial to shallow marine environments (WSRC 1993e).

Figure 4-11 shows the current aquifer/aquitard terminology at the SRS. Aquifers, in ascending order, include the McQueen Branch, the Crouch Branch, and the Steed Pond. For comparison, the figure also includes the corresponding aquifer terminology used on the Georgia side of the Savannah River. These include the Midville, Dublin, and Floridan aquifer systems. In addition, the three aquifers are separated by confining layers which include, in ascending order, the Appleton, Allendale, and Meyers Branch confining systems (WSRC 1993e).

4.8.2.2 Groundwater Flow. Excellent quality groundwater is abundant in this region of South Carolina from many local aquifer units. As a result, the South Carolina Department of Health and Environmental Control has classified all aquifers in the state as Class GB (South Carolina 1976), or U.S. Environmental Protection Agency (EPA) Class II, meaning that the aquifers can provide



Note: Not to scale

SFIG 0411

Figure 4-11. Comparison of lithostratigraphy and hydrostratigraphy for the SRS region.

resource-quality water, but are not the sole source of supply (South Carolina Class GA or EPA Class I aquifers) (DOE 1991b).

The main source of recharge to the vadose zone is rainfall. The annual precipitation at the SRS is 48 inches (121.9 centimeters), with an estimated 16 inches (41 centimeters) designated as surface recharge at the center of the SRS, in bare and grass-covered areas (WSRC 1993e). The direction of groundwater flow in the vadose zone is predominantly downward. However, given the lenses of silt and clay that exist, there is significant lateral spread in some areas. In general, the vadose zone thickness ranges from approximately 130 feet (40 meters) in the northernmost portion of the SRS to 0 feet where the water table intersects wetlands, streams, or creeks.

The following discussion of groundwater flow in the Coastal Plain hydrogeologic province begins with the deepest aquifers at the SRS and proceeds to shallower units. It does not address flow in the confining units because few hydraulic head measurements are available for these units and, to a good approximation, flow in aquitards is limited predominantly to vertical flow between aquifer units. The Midville or McQueen Branch aquifer (which has also been called the Middendorf, the Lower Cretaceous, the Tuscaloosa, and Aquifer IA) is highly transmissive and, therefore, serves in part as the production aquifer for much of the SRS. This aquifer flows horizontally, predominantly toward the Savannah River. In the past, groundwater production wells at the SRS were screened in both the Midville (McQueen Branch) and Dublin (Crouch Branch) aquifers. In 1985 DOE committed to the South Carolina Department of Health and Environmental Control to complete production wells only in the McQueen Branch aquifer to minimize the potential for contamination to reach such wells and spread in the deeper aquifers.

Flow in the Dublin or Crouch Branch aquifer (which has also been called the Black Creek, the Tuscaloosa, the Upper Cretaceous, and Aquifer IB) is more complicated than flow in the deeper McQueen Branch aquifer because of the apparent communication with Upper Three Runs Creek on the SRS. Nonetheless, horizontal flow in the Dublin (Crouch Branch) aquifer is predominantly toward the Savannah River. However, there is an upward vertical flow component near the river and Upper Three Runs Creek. Recharge to the Dublin-Midville aquifer system occurs in areas exposed at the ground surface near the Fall Line (see Figure 4-3).

Horizontal flow in the Gordon aquifer (previously called the Congaree, the Tertiary, and Aquifer II) is toward Upper Three Runs Creek and the Savannah River, depending on the area of the

SRS. Both the river and Upper Three Runs Creek intercept this aquifer. The Gordon aquifer receives most of its recharge from groundwater that originates on the SRS.

Previous SRS studies have called the Upper Three Runs aquifer the "water table aquifer"; others have defined it as both the Barnwell/McBean and water table aquifers in the central portion of the SRS where those aquifers were thought to be separated by a "tan clay." The Upper Three Runs aquifer is the shallowest aquifer at the SRS. The horizontal groundwater flow is generally toward the nearest surface-water feature that is in communication with the water table. Most SRS streams, except Tims Branch in the northeastern part of the Site, are in communication with the water table. Tims Branch is a "losing stream," meaning it provides, or "loses," water to the Upper Three Runs aquifer. However, the Upper Three Runs aquifer receives most of its recharge from precipitation. The Upper Three Runs aquifer is not a source of domestic or production water on the SRS because the lower aquifers provide a more abundant supply of higher quality water (WSRC 1993e).

4.8.2.3 Groundwater Quality. The quality of groundwater in the principal hydrologic systems beneath the SRS depends on both the source of the water and the inorganic and biochemical reactions that take place along its flowpath. Quality is strongly influenced by the chemical composition and mineralogy of the enclosing geologic materials (WSRC 1993e).

In general, the quality of the groundwater in the Coastal Plain sediments at the SRS and the surrounding areas is suitable for most domestic and industrial purposes. The waters have low concentrations of total dissolved solids (TDS), ranging from less than 10 milligrams per liter to about 150 to 200 milligrams per liter. The pH values range from 4.9 to 7.7 (where the groundwater is in contact with limestone). Much of the groundwater is corrosive to metal surfaces due to its low solids content and frequently low pH values. High dissolved iron concentrations can also be of concern in some groundwater units. The SRS uses degasification and filtration processes to raise the pH and remove iron in domestic water supplies where necessary (WSRC 1993e).

Table 4-12 summarizes groundwater quality data from 85 existing waste sites on the SRS compared to drinking water standards; Table 4-13 lists similar information for selected radiological constituents. The data in these tables are from ongoing monitoring programs on the Site. EPA-accepted methods and guidelines for sampling and analysis are an integral part of this monitoring program. Several of the facilities discussed below have state-approved sampling and analysis plans.

Table 4-12. Representative groundwater quality data for nonradioactive constituents from the Savannah River Site.[a]

Parameter (Unit)	Standard	Maximum Value
Alkalinity (as $CaCO_3$) (mg/L)	100	1,360[b]
pH (pH units)	8.5[c]	13[b]
Antimony (mg/L)	0.005	0.013
Arsenic (mg/L)	0.05	0.1
Beryllium (mg/L)	0.011[d]	0.0043
Cadmium (mg/L)	0.005[c]	0.34
Chromium (mg/L)	0.1[c]	0.82
Mercury (mg/L)	0.002[c]	0.12
Lead (mg/L)	0.015[e]	1.0
Nitrate-N (mg/L)	10[c]	278[b]
Sulfate (mg/L)	400[c]	73,500[b]
Pentachlorophenol (mg/L)	0.001[c]	0.0032
Lindane (mg/L)	0.0002[c]	0.00048
Carbon tetrachloride (mg/L)	0.005	0.43
1,2-Dichloroethane (mg/L)	0.005[c]	0.27
1,1,1-Trichloroethane (mg/L)	0.2[c]	0.21
1,1-Dichloroethylene (mg/L)	0.007[c]	0.15
Trichlorethylene (mg/L)	0.005[c]	147
Tetrachloroethylene (mg/L)	0.005[c]	101

a. Data compiled from 85 existing wastes sites (Arnett et al. 1993).
b. The elevated values for alkalinity and pH might be due to faulty well installation; the elevated sulfate and nitrate values might be due to acid spills near wells.
c. National secondary drinking water regulations (40 CFR Part 143), 1991.
d. National primary drinking water regulations (40 CFR Part 141).
e. Action level at which providers of public drinking water apply treatment technique to reduce lead levels (40 CFR Part 143).

Table 4-13. Representative groundwater data for radioactive constituents from the Savannah River Site (pCi/liter).[a]

Constituent	Standard[b]	Maximum Concentration
Gross alpha	15	2,700
Nonvolatile beta	50	19,000
Tritium	20,000	1.8×10^8
Cesium-137	200	980
Cobalt-60	100	290
Iodine-129	1	72
Ruthenium-106	30	170
Total radium (radium-226 and radium-228)	5	50
Strontium-90	8	5,300

a. Source: Arnett et al. (1993).
b. National Primary Drinking Water Regulations, Radionuclides, 40 CFR Part 141, 56 FR 33052.

The shallow aquifers beneath 5 to 10 percent of the SRS have been contaminated by industrial solvents, metals, tritium, or other constituents used or generated on the Site. Figure 4-12 shows the locations of facilities where the SRS monitors groundwater and areas with constituents that exceeded drinking water standards in 1992; the concentrations shown on Figure 4-12 represent the maximum data from one monitoring well on at least one occasion at a given area. Contamination is limited to the shallow aquifers, with one exception (see next paragraph). Most contaminated groundwater at the SRS is beneath a few facilities; contaminants reflect the operations and chemical processes those facilities perform. For example, contaminants in the groundwater beneath A- and M-Areas include chlorinated volatile organics, radionuclides, metals, and nitrate. At F- and H-Areas, contaminants in the groundwater include tritium and other radionuclides, metals, nitrate, chlorinated volatile organics at values much smaller than those found at A- and M-Areas, and sulfate. The groundwater beneath the Sanitary Landfill contains chlorinated volatile organics, radionuclides, and metals. The groundwater beneath all the reactor areas except R-Area contains tritium, other nuclides, metals, and chlorinated volatile organics. At R-Area, groundwater contaminants include radionuclides and cadmium. The groundwater beneath D-Area contains metals, radionuclides, sulfate, and chlorinated volatile organics. At TNX-Area, the groundwater contains chlorinated volatile organics, radionuclides, and nitrate (Arnett

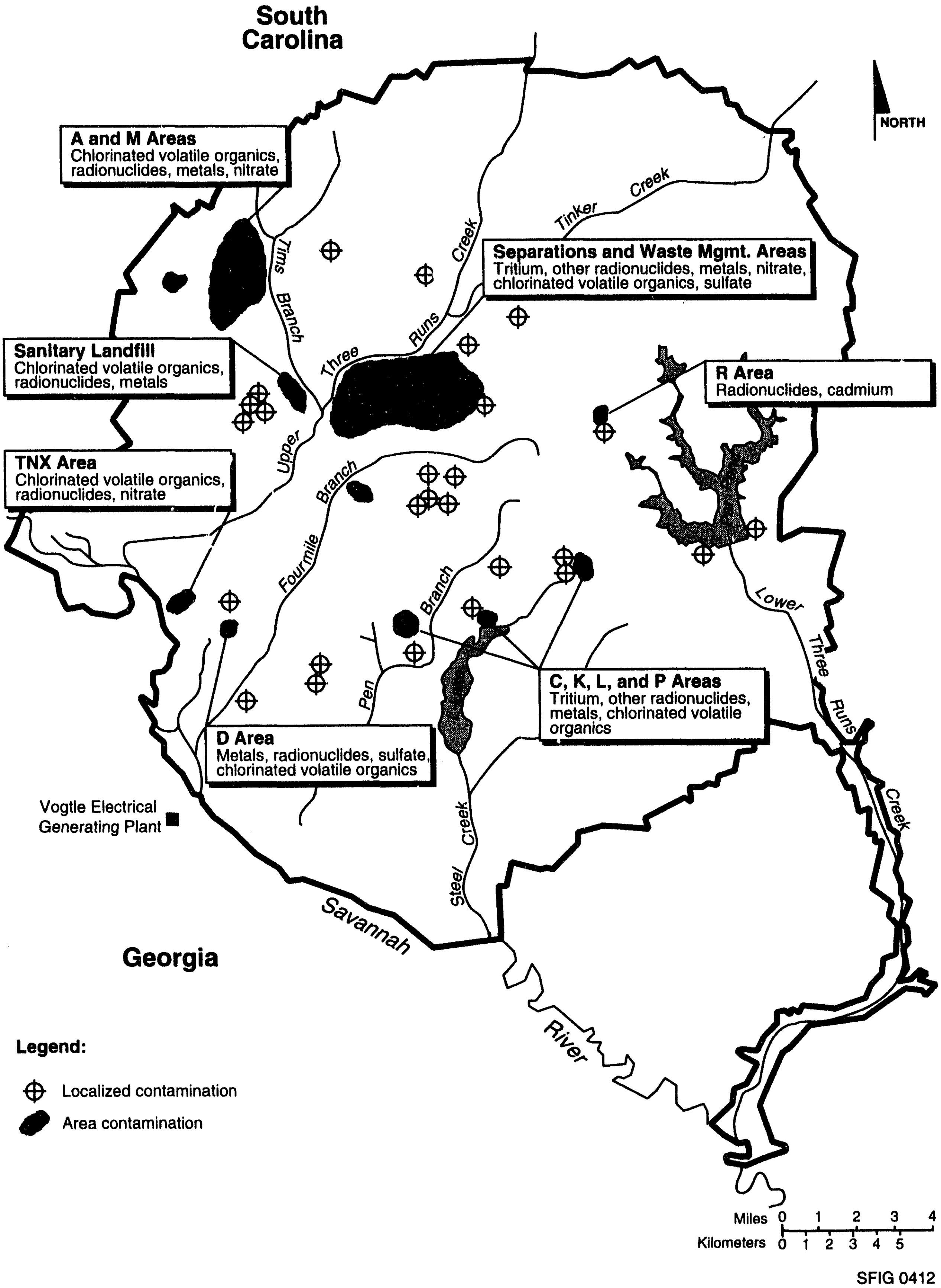


Figure 4-12. Groundwater contamination at the Savannah River Site.

et al. 1993). None of these cases indicated the presence of groundwater contamination beyond Site boundaries. With the ongoing and expanding "pump and treat" system at the A/M-Area (Figure 4-12), concentrations in the volatile organic compound plume are likely to decrease with time.

Contamination of groundwater in a drinking water aquifer has been found in only one localized area beneath the site. In the early 1980s, SRS monitors found low concentrations of trichloroethylene (11.7 microgram per liter) in water from one production well (53A) completed to the Dublin-Midville Aquifer System (formerly called the Tuscaloosa Formation) in M-Area. The monitors found the contamination only at 430 and 480 feet (131 and 146 meters) in this well, which is 670 feet (204 meters) deep. The well is screened intermittently from 387 feet (118 meters) to the bottom. DOE concluded that the contamination is probably migrating down the outside well casing from soils near the surface that are contaminated with trichloroethylene. This contaminated water enters the well through screens set in the Dublin-Midville System (Du Pont 1983). Continued monitoring since the initial discovery has not found contamination in other wells in the area.

4.8.2.4 Groundwater Use. The McQueen Branch aquifer, which becomes shallower toward the Fall Line, forms the base for most municipal and industrial water supplies in Aiken County. Toward the coast, in Allendale and Barnwell Counties, this aquifer exists at increasingly greater depths. As a consequence, the shallower Gordon aquifer supplies some municipal, industrial, and agricultural users. The Gordon and Upper Three Runs Creek aquifers are the primary sources for domestic water supplies in the vicinity of the SRS.

DOE has identified 56 major municipal, industrial, and agricultural groundwater users within 20 miles (32 kilometers) of the center of the SRS (DOE 1987a). The total pumpage for these users is about 36 million gallons (135,000 cubic meters) per day.

4.9 Ecological Resources

The U.S. Government acquired the SRS in 1951. At that time, the Site was approximately two-thirds forested and one-third cropland and pasture (Dukes 1984). At present, more than 90 percent of the SRS is forested. An extensive forest management program conducted by the Savannah River Forest Station, which is operated by the U.S. Forest Service, has converted many pastures and croplands to pine plantations. With the exception of the SRS production and support areas, natural succession has reclaimed previously disturbed areas. Table 4-14 lists SRS land cover, other than the land used for nuclear reactors and support facilities.

The SRS is important to maintaining the biodiversity of the region. Satellite imagery of the Site shows a circle of wooded habitat within a matrix of cleared uplands and narrow forested riparian corridors. The SRS provides more than 734 square kilometers (181,000 acres) of contiguous forested cover broken only by unpaved secondary roads, transmission line corridors in various stages of succession, and a few paved primary roads. Carolina bays, the Savannah River swamp, and several relatively intact longleaf pine-wiregrass communities provide important contributions to the biodiversity of the SRS and of the entire region.

F- and H-Areas, located near the center of the SRS and approximately 1.6 kilometers (1 mile) southeast of Upper Three Runs Creek, are heavily industrialized with little natural vegetation remaining inside the fenced areas. These areas are dominated by buildings, paved parking lots, gravelled construction areas, and laydown yards. While some grassed areas occur around the administration buildings and some vegetation is present along the ditches that drain the area, the majority of the site contains no vegetation. Wildlife is absent except for occasional crows (*Corvus brachyrhynchos*) and nesting barn swallows (*Hirundo rustica*) around the buildings.

Figure 2-3 shows the location of a representative host site at the SRS for potential spent nuclear fuel activities. F- and H-Areas (and developed areas immediately adjacent to them) would house most spent nuclear fuel management facilities, while the undeveloped area south and east of H-Area would be used for the construction of new facilities that F- and H-Areas could not accommodate. The undeveloped area, which was 98 percent cleared fields in 1951, is now almost completely forested, for the most part with 5- to 40-year-old upland pine stands that are actively managed by the Savannah River Forest Station. Most of these stands are loblolly pine (*Pinus taeda*), but there are small stands of slash pine (*P. elliottii*), upland hardwoods (predominantly oaks and hickories), and bottomland hardwoods (most commonly sweetgum, *Liquidambar styraciflua*, and yellow poplar, *Liriodendron tulipifera*) associated with two small Carolina bays located south of H-Area. The area south of H-Area lies in the Fourmile Branch watershed, while the area east of H-Area is in the McQueen Branch (a tributary of Upper Three Runs Creek) watershed. Neither area is likely to contain any threatened or endangered species or their habitats.

Table 4-14. Land cover of undeveloped areas on the Savannah River Site.[a]

Land cover types	Square kilometers[b]	Percent of total
Longleaf pine	150	20
Loblolly pine	258	35
Slash pine	117	16
Mixed pine/hardwood	23	3
Upland hardwood	20	3
Bottomland hardwood	117	16
Savannah River swamp	49	7
Total	734	100.0

a. Source: USDA (1991a).
b. To convert square kilometers to acres, multiply by 247.1.

The general area of the representative host site contains suitable habitat for white-tailed deer and feral hogs as well as other faunal species common to the mixed pine/hardwood forests of South Carolina. Additional wildlife species found in the area include gray squirrel (*Sciurus carolinensis*), fox squirrel (*S. niger*), wild turkey (*Meleagris gallopovo*), cottontail rabbit (*Sylvilagus floridanus*), raccoon (*Procyon lotor*), bobcat (*Felix rufus*), and gray fox (*Urocyon cinereoargenteus*).

4.9.1 Terrestrial Ecology

The SRS is near the transition area between the oak-hickory-pine forest and the southern mixed forest. As a consequence, species typical of both associations occur (Dukes 1984). In addition, farming, fire, soil features, and topography have strongly influenced existing SRS vegetation patterns.

A variety of vascular plant communities occurs in the upland areas (Dukes 1984). Typically, scrub oak communities occur on the drier, sandier areas. Longleaf pine (*Pinus palustrus*), turkey oak (*Quercus laevis*), bluejack oak (*Q. incana*), blackjack oak (*Q. marilandica*), and dwarf post oak (*Q. margaretta*) dominate these communities, which typically have understories of wire grass (*Aristida stricta*) and huckleberry (*Vaccinium* sp.). Oak-hickory communities occur on more fertile, dry uplands; characteristic species are white oak (*Q. alba*), post oak (*Q. stellata*), southern red oak (*Q. falcata*), mockernut hickory (*Carya tomentosa*), pignut hickory (*C. glabra*), and loblolly pine, with an understory of sparkleberry (*Vaccinium arboreum*), holly (*Ilex* sp.), greenbriar (*Smilax* sp.), and poison ivy (*Rhus radicans*).

The removal of human residents in 1951 and the subsequent restoration of forest cover has provided the wildlife of the SRS with excellent habitat. Furbearers such as gray fox, raccoon, opossum (*Didelphis virginiana*), bobcat, beaver (*Castor canadensis*), and otter (*Lutra canadensis*) are relatively common throughout the Site. Game species such as gray squirrel and fox squirrel, white-tailed deer (*Odocoileus virginianus*), cottontail rabbit, and wild turkey are also common. The Savannah River Ecology Laboratory has made extensive studies of reptile and amphibian use of the wetlands and adjacent uplands of the SRS.

DOE allows carefully regulated public hunting for white-tailed deer and feral hogs (*Sus scrofa*) on most of the SRS to reduce the incidence of animal/vehicle collisions and maintain healthy populations within the carrying capacity of the range. SRS personnel monitor all animals removed from the Site for contamination before releasing them to the hunters (WSRC 1992a).

Before releasing any animal to a hunter, SRS technicians perform field analyses for cesium-137 at the hunt site. In 1992, hunters collected 1,519 deer and 168 hogs. The maximum 1992 cesium-137 field measurement for deer was 22.4 picocuries per gram; the average was 6.4 picocuries per gram (Arnett et al. 1993). For hogs, the maximum value was 22.9 picocuries per gram and the average was 3.5 picocuries per gram. The field technicians determine estimated doses from consumption of the venison and pork and make this information available to the hunters.

In 1992, the estimated maximum dose received by a hunter was 49 millirem per year. The basis for this unique hypothetical maximum dose, which was for a hunter who harvested eight deer and one hog, is the assumption that the hunter consumed the entire edible portion of each animal. An additional hypothetical model involved a hunter whose total meat consumption for the year consisted of SRS deer [81 kilograms (179 pounds) per year] (Hamby 1991). Based on these low-probability assumptions and on the average concentration of cesium-137 (6.4 picocuries in deer harvested on the SRS), the estimated potential maximum dose from this pathway is 26 millirem; this is 26 percent of the annual 100-millirem DOE Derived Concentration Guide. Although a large percentage of this hypothetical dose is probably due to cesium-137 from worldwide fallout, the estimated total contains this background cesium-137 for conservatism.

4.9.2 Wetlands

The SRS has extensive, widely distributed wetlands, most of which are associated with floodplains, creeks, and impoundments. In addition, approximately 200 Carolina bays occur on the Site (Shields et al. 1982; Schalles et al. 1989).

The southwestern SRS boundary adjoins the Savannah River for approximately 32 kilometers (20 miles). The river floodplain supports an extensive swamp, covering about 49 square kilometers (12,148 acres) of the Site; a natural levee separates the swamp from the river. Timber was cut in the swamp in the late 1800s. At present, the swamp forest consists of second-growth bald cypress (*Taxodium distichum*), black gum (*Nyssa sylvatica*), and other hardwood species (Workman and McLeod 1990; USDA 1991a).

Five major streams drain the SRS and eventually flow into the Savannah River. Each stream has floodplains characterized by bottomland hardwood forests or scrub-shrub wetlands in varying stages of succession. Dominant species include red maple (*Acer rubrum*), box elder (*A. negundo*), bald cypress, water tupelo (*Nyssa aquatica*), sweetgum, and black willow (*Salix nigra*) (Workman and McLeod 1990).

Carolina bays are unique wetland features of the southeastern United States. They are islands of wetland habitat dispersed throughout the uplands of the SRS. The approximately 200 bays on the Site exhibit extremely variable hydrology and a range of plant communities from herbaceous marsh to forested wetland (Shields et al. 1982; Schalles et al. 1989). SRS scientists have studied Carolina bay ecology extensively, particularly in relation to the construction of the Defense Waste Processing Facility (DWPF; SREL 1980).

4.9.3 Aquatic Ecology

The aquatic resources of the SRS have been the subject of intensive study for more than 30 years. Research has focused on the flora and fauna of the Savannah River and the five tributaries of the river that drain the Site. Section 4.8.1.1 describes those portions of the aquatic systems that spent nuclear fuel management activities could affect. In addition, several monographs (Patrick et al. 1967; Dahlberg and Scott 1971; Bennett and McFarlane 1983), the eight-volume Comprehensive Cooling Water Study (Du Pont 1987), and three EISs (DOE 1984; DOE 1987b; DOE 1990) that evaluated operations of SRS production reactors describe the aquatic biota and aquatic systems of the SRS.

4.9.4 Threatened and Endangered Species

Threatened, Endangered, and Candidate Plant and Animal Species of the Savannah River Site (HNUS 1992b) describes threatened, endangered, and candidate plant and animal species that are known to occur or that might occur on the SRS. Table 4-15 lists these species.

The following Federally listed endangered animals are known to occur on the SRS or in the Savannah River adjacent to the Site: the red-cockaded woodpecker (*Picoides borealis*), the southern bald eagle (*Haliaeetus leucocephalus*), the wood stork (*Mycteria americana*), and the shortnose sturgeon (*Acipenser brevirostrum*) (HNUS 1992b). Researchers have found one Federally listed endangered plant species, the smooth coneflower (*Echinacea laevigata*), on the Site, several Federally listed Category 2 species, and several state listed species (Knox and Sharitz 1990). At present, the SRS is implementing strategies for the protection of these species.

F- and H-Areas and the representative host site contain no habitat suitable for any of the Federally listed threatened or endangered species found on the SRS. The Southern bald eagle and the wood stork feed and nest near wetlands, streams, and reservoirs, and thus would not be attracted to the host site, a densely forested upland area. Shortnose sturgeon, typically residents of large coastal rivers and estuaries, have never been collected in Fourmile Branch or any of the tributaries of the Savannah River that drain the SRS.

Red-cockaded woodpeckers prefer open pine forests with mature trees (older than 80 years) for foraging and nesting. The pines of the undeveloped host site are 5 to 40 years old, thus red-cockaded woodpeckers probably would not forage or nest in the area.

The *Red-Cockaded Woodpecker Standards and Guidelines, Savannah River Site* (USDA 1991b) describes the SRS management strategy for the red-cockaded woodpecker. The most significant element of this management strategy is the conversion of slash (and some loblolly) pine in a designated red-cockaded woodpecker management area to longleaf pine, with a harvest rotation of 120 years.

Table 4-15. Threatened, endangered, and candidate plant and animal species of the SRS.

Common Name (Scientific Name)	Status
Animals	
Rafinesques (= Southeastern) big-eared bat (*Plecotus rafinesquii*)	FC2
Loggerhead Shrike (*Lanius ludovicianus*)	FC2
Bachman's sparrow (*Aimophila aestivalis*)	FC2
Carolina crawfish (= Gopher) frog (*Rana areolata capito*)	FC2
Southern hognose snake (*Heterodon simus*)	FC2
Northern pine snake (*Pituophis melanoleucus melanoleucus*)	FC2
Bald eagle (*Haliaeetus leucocephalus*)	E
Wood stork (*Mycteria americana*)	E
Red-cockaded woodpecker (*Picoides borealis*)	E
American alligator (*Alligator mississippiensis*)	T/SA
Shortnose sturgeon (*Accipenser brevirostrum*)	E
Plants	
Smooth coneflower (*Echinacea laevigata*)	E
Bog spice bush (*Lindera subcoriacea*)	FC2
Boykin's lobelia (*Lobelia boykinii*)	FC2
Loose watermilfoil (*Myriophyllum laxum*)	FC2
Nestronia (*Nestronia umbellula*)	FC2
Awned meadowbeauty (*Rhexia aristosa*)	FC2

Key: E = Federal endangered species.
T/SA = Threatened due to Similarity of Appearance.
FC2 = Under review (a candidate species) for listing by the Federal government.

4.10 Noise

The major noise sources at the SRS occur primarily in developed operational areas and include various facilities, equipment, and machines (e.g., cooling towers, transformers, engines, pumps, boilers, steam vents, paging systems, construction and materials-handling equipment, and vehicles). Major noise sources outside the operational areas consist primarily of vehicles and railroad operations. Previous studies have assessed noise impacts of existing SRS operational activities (NUS 1991b; DOE 1991b; DOE 1990; DOE 1993a). These studies concluded that, because of the remote locations of the SRS operational areas, there are no known conditions associated with existing onsite noise sources that adversely affect individuals at offsite locations. Some disturbance of wildlife activities might occur on the SRS as a result of operational and construction activities.

Existing SRS-related noise sources of importance to the public are those resulting from the transportation of people and materials to and from the Site. These sources include trucks, private vehicles, helicopters, and freight trains. In addition, a portion of the air cargo and business travel using commercial air transport through the airports at Augusta, Georgia, and Columbia, South Carolina, are attributable to SRS operations.

The States of Georgia and South Carolina and the counties in which the SRS is located have not established any regulations that specify acceptable community noise levels with the exception of Aiken County. A provision of the Aiken County Nuisance Ordinance limits daytime and nighttime noise by frequency band (Aiken County 1991).

During a normal week in 1995, about 20,000 employees are likely to travel to the SRS each day in private vehicles from surrounding communities. Both government-owned and private trucks pick up and deliver materials at the Site. Most private vehicles and trucks traveling to and from the Site each day use South Carolina Highways (SC) 125 and 19. The contribution of SRS operations to traffic volumes along SC 125 and SC 19, especially during peak traffic periods, affects noise levels through the towns of New Ellenton and Jackson and the City of Aiken.

Noise measurements taken during 1989 and 1990 along SC 125 in the Town of Jackson at a point about 15 meters (50 feet) from the roadway indicate that the 1-hour equivalent sound level from traffic ranged from 48 to 72 decibels (A-weighted). The estimated day/night average sound level along this route was 66 decibels for summer and 69 decibels for winter. Similarly, noise measurements along SC 19 in the town of New Ellenton at a point about 15 meters (50 feet) from the roadway indicate that the 1-hour equivalent sound level from traffic ranged from 53 to 71 decibels. The estimated day/night average sound level along this route was 68 decibels for summer and 67 decibels for winter (NUS 1990). Employment at the SRS has increased slightly since 1989, potentially causing small increases in traffic noise, especially during peak traffic periods (approximately between 6:30 and 8:30 a.m. and between 3:30 and 5:30 p.m., corresponding to the major shift changes). Because some residences and at least two schools are within 100 to 200 feet of these routes, some annoyance to members of the public residing along these highways might occur based on the relationship between the day/night average sound level and the "percent highly annoyed" (Schultz 1978; Fidell et al. 1989; FICON 1992).

As discussed in Section 4.11, approximately 13 trains per day pass through the SRS on the CSX line, with 5 trains per week delivering shipments to the SRS (Burns 1993; Graves 1993). Noise sources from rail transport include diesel engines, wheel-track contact, and whistle-warnings at rail crossings.

4.11 Traffic and Transportation

4.11.1 Regional Infrastructure

The SRS is surrounded by a system of Interstate highways, U.S. highways, state highways, and railroads. The regional transportation networks service the four South Carolina counties (Aiken, Allendale, Bamberg, and Barnwell) and two Georgia counties (Columbia and Richmond) that generate about 90 percent of SRS commuter traffic (HNUS 1992a). Two major railroads - CSX Transportation and Norfolk Southern Corporation - also serve the SRS vicinity. Although barge traffic is possible on the Savannah River, neither the SRS nor commercial shippers normally use barges. Figure 4-13 shows the regional transportation infrastructure.

4.11.1.1 Regional Roads. Two Interstate highways serve the SRS area. Interstate 20 (I-20) provides a primary east-west corridor and I-520 links I-20 with parts of Augusta, Georgia. U.S. Highways 1 and 25 are principal north-south routes and U.S. 78 provides east-west connections. Several other highways - U.S. 221, U.S. 301, U.S. 321, and U.S. 601 - provide additional transport routes in the region.

Several state routes provide direct access to the SRS. Running northwest/southeast is SC 125. Access to the Site is provided from the north by SC 19, from the northeast by SC 39, and from the east by SC 64.

U.S. 278 bisects the northern part of the SRS and is available to public access without restriction. The SRS maintains barricades at site entries and exits on SC 125 to control public access if necessary, although it is generally open to unrestricted public travel. The public also has direct access to Site Road 1. All other site roads have restricted access.

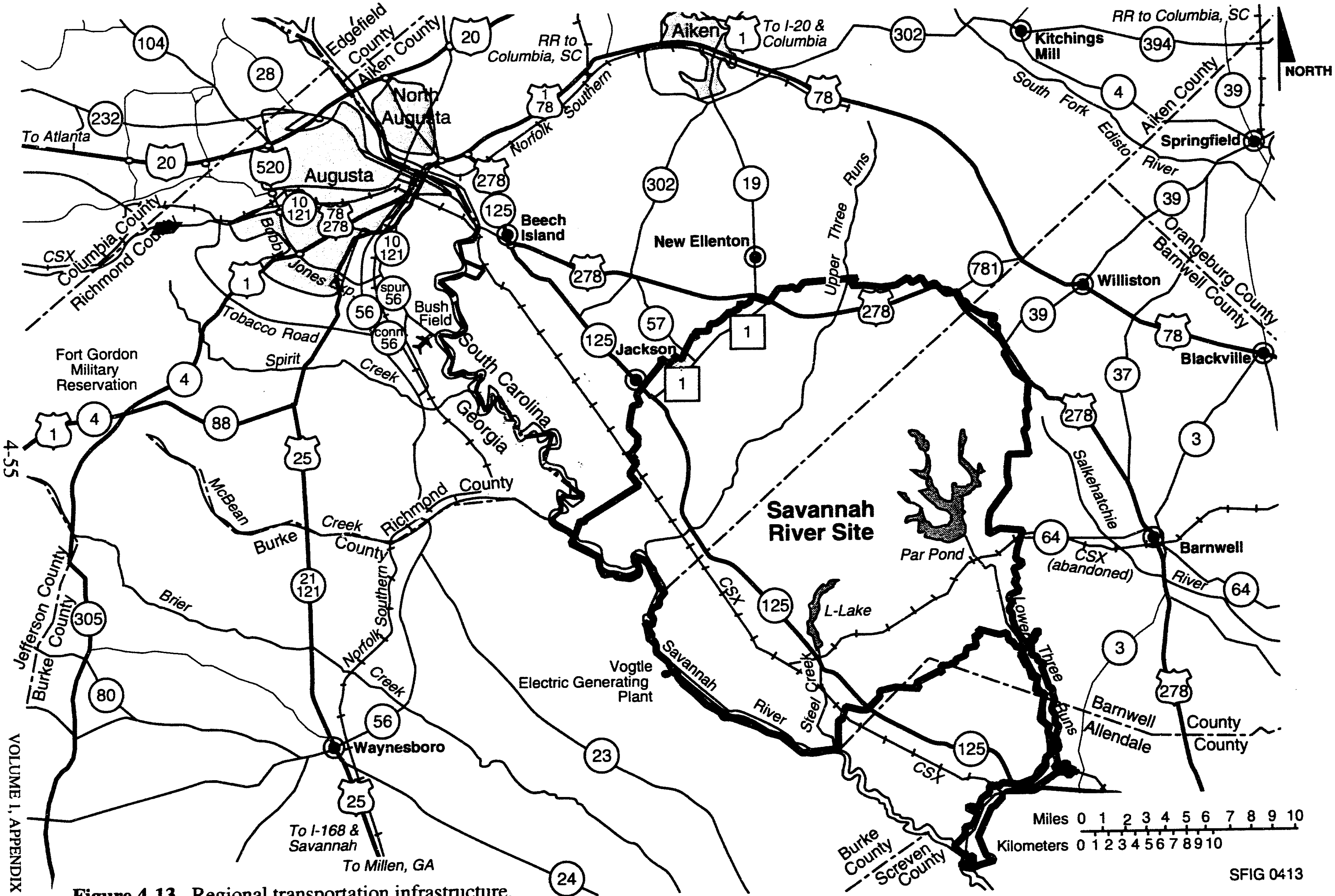


Figure 4-13. Regional transportation infrastructure.

4.11.1.2 Regional Railroads. Norfolk Southern serves Augusta and Savannah, Georgia, as well as Columbia and Charleston, South Carolina. CSX serves the same locations and the SRS.

4.11.2 SRS Infrastructure

The SRS transportation infrastructure consists of more than 143 miles (230 kilometers) of primary roads, 1,200 miles (1,931 kilometers) of unpaved secondary roads, and 103 kilometers (64 miles) of railroad track (WSRC 1993d). These roads and railroads provide connections among the various SRS facilities and to offsite transportation linkages. Figure 4-14 shows the SRS network of primary roadways and access points. Figure 4-15 shows the SRS railway system.

4.11.2.1 SRS Roads. Two major public highways traverse the Site: SC 125 and U.S. 278. SC 125 connects Allendale, South Carolina, to Augusta, Georgia, by crossing the Site in a northwest-to-southeast direction. U.S. 278 also connects Augusta and Allendale, but its route approximately follows the northern and eastern SRS boundaries.

Ten barricades around the Site limit access from public roads. Five barricades limit SRS access from SC 125; three limit access from SC 19, SC 39, and SC 64; and two limit access from the public areas of the administrative complex near the northern SRS boundary (A-Area).

In general, the primary SRS roadways are in good condition and are smooth and free from potholes. Typically, wide, firm shoulders border roads that are either straight or have wide gradual turns. Intersections are well marked for both traffic and safety identification and are sufficiently cleared of trees and brush that might obstruct a driver's view of oncoming traffic. Railings along the side of the roadways offer protection at appropriate locations from dropoffs or other hazards. In general, the roadways are lighted only at gate areas and near major facilities.

The SRS has two overpasses, one at the cloverleaf intersection of Roads 2 and C, and the other where SC 125 overpasses the CSX railroad tracks in the southern part of the Site. The 60 bridges on the Site have been inspected and evaluated for safe loading, with some bridges rated as high as 200 tons (181 metric tons) under controlled conditions. The steepest roadway gradient is on Road C at the east bank of Upper Three Runs Creek, where the road drops more than 100 feet (30 meters) in about 0.25 miles (0.4 kilometer). At the base of the dropoff is a bridge over the creek and an immediate turn in the road. This area presents a relatively hazardous roadway condition.

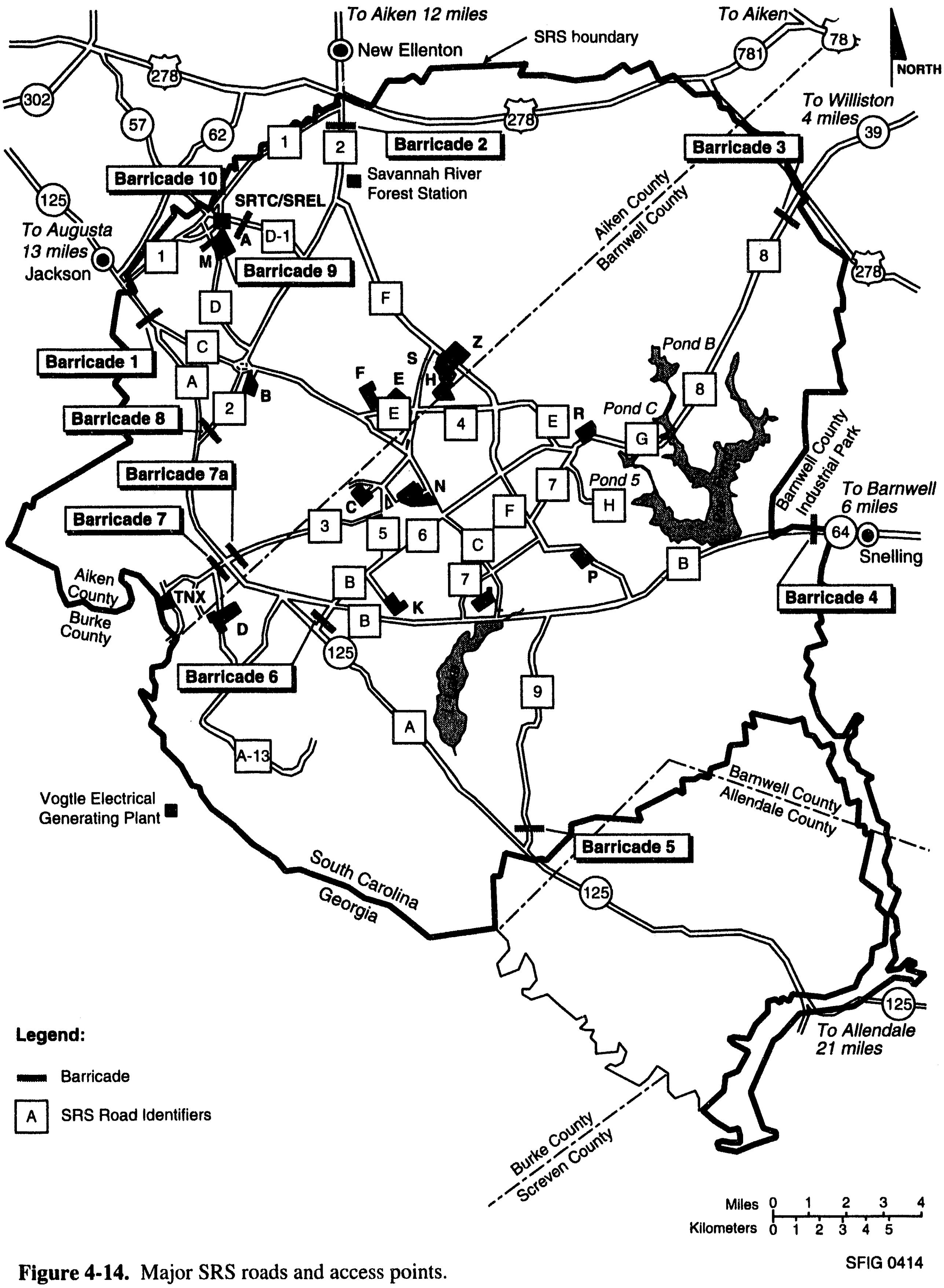


Figure 4-14. Major SRS roads and access points.

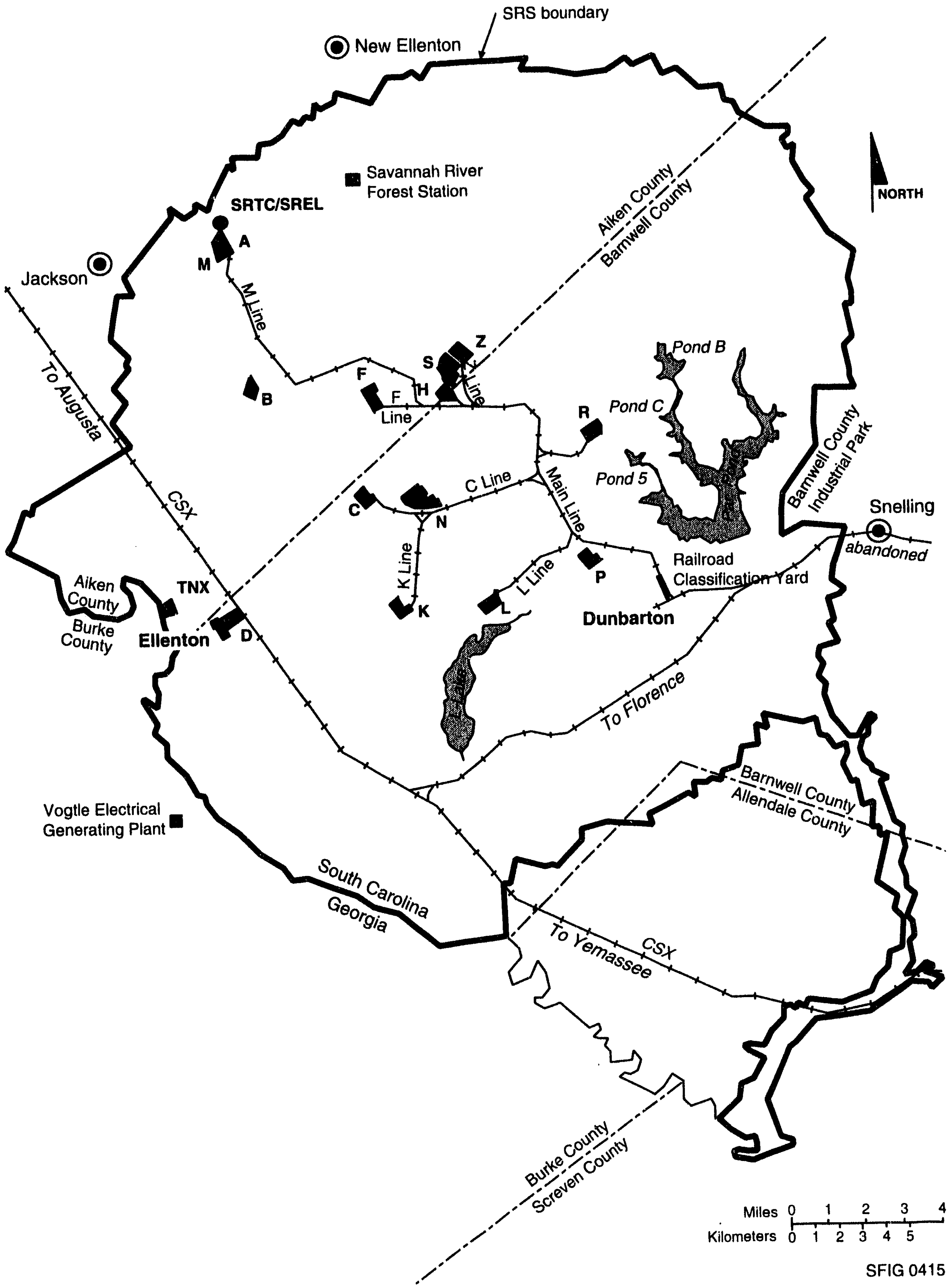


Figure 4-15. SRS railroad lines.

In general, heavy traffic occurs early in the morning and late in the afternoon when workers from surrounding communities commute to and from the Site. During working hours, official vehicles and logging trucks constitute most of the traffic. At any time, as many as 60 logging trucks, which can impede traffic, might be operating on the Site, with an annual average of about 25 trucks per day. Table 4-16 provides data on traffic counts for various roads and access points around the SRS.

4.11.2.2 SRS Railroads. Railroads on the Site include both CSX tracks and SRS rolling stock and tracks. Two routes of the CSX distribution system run through the Site: a line between Florence, South Carolina, and Augusta, Georgia, and a line between Yemassee, South Carolina, and Augusta. The two lines join on the Site near the L-Lake dam (Figure 4-15). Early in 1989 CSX discontinued service on the line from the SRS junction to Florence.

The 64 miles (103 kilometers) of SRS railroads are well maintained. The rails and crossties are in good condition, and the track lines are clear of vegetation and debris. Significant clear areas border the tracks on both sides. Intersections of railroads and roadways are marked by railroad crossing signs with lights where appropriate.

The SRS rail classification yard is east of P-Reactor. This eight-track facility sorts and redirects rail cars. Deliveries of SRS shipments occur at two onsite rail stations at the former towns of Ellenton and Dunbarton. From these stations, an SRS engine moves the railcars to the appropriate receiving facility. The Ellenton station, which is on the main Augusta-Yemassee line, is the preferred delivery point. The Dunbarton station, which is on the discontinued portion of the Augusta-Florence line, receives less use.

Under normal conditions, about 13 trains per day use the CSX tracks through the SRS (Burns 1993). The number of shipments to the SRS varies from week to week but currently averages about five trains per week (Graves 1993). Most shipments contain coal; the remaining shipments consist primarily of tank cars, typically carrying sulfuric acid or sodium hydroxide. Occasionally a shipment consists of a flat car carrying a heavy piece of equipment.

4.12 Occupational and Public Radiological Health and Safety

The sources of radiation exposure to individuals consist of natural background radiation from cosmic, terrestrial, and internal body sources; radiation from medical diagnostic and therapeutic

Table 4-16. SRS traffic counts - major roads.[a]

Measurement point	Date	Direction	Day Total	Peak[b]	Peak time[c]	Average speed (mph)[d]
Road 2 between Roads C and D	2-23-93	East	3,031	800	1530	47
	4-21-93	West	3,075	864	0630	NA[e]
Road 4 between Roads E and C	12-9-92	East	1,624	352	1530	NA
	12-9-92	West	1,553	306	0615	NA
Road 8 at Pond C	2-23-92	East	634	274	1530	58
	2-23-92	West	662	331	0615	56
Road C between landfill and Road 2	12-16-92	North	6,931	2,435	1530	53
	12-16-92	South	6,873	2,701	0630	58
Road C north of Road 7	1-20-93	North	742	288	0630	53
	1-20-93	South	763	223	1530	54
Road D	9-29-93	North	1,779	218	1500	43
	9-29-93	South	1,813	220	0845	52
Road E at E-Area	8-25-93	North	3,099	669	1530	35
	8-25-93	South	3,054	804	0630	38
Road F at Upper Three Runs Creek	2-2-93	North	3,239	1,438	1530	53
	2-2-93	South	3,192	1,483	0630	51
H-Area Exit	12-2-92	Outbound	2,181	406	1530	12

a. Source: Swygert (1993).
b. Number of vehicles in peak hour.
c. Start of peak hour.
d. mph = miles per hour; to convert to kilometers per hour multiply by 1.6093.
e. NA = data not available.

practices; and radiation from manmade sources, including consumer and industrial products, nuclear facilities, and weapons test fallout.

All radiation doses discussed in this document are effective dose equivalents (i.e., organ dose equivalents weighted for biological effect and summed to yield a whole-body dose equivalent with the same risk as irradiation of individual organs) as defined by the International Commission on Radiological Protection, Publication 26 (ICRP 1977), unless specifically identified otherwise (e.g., thyroid dose, bone dose).

Natural background radiation contributes about 83 percent of the annual dose of 380 millirem received by an average member of the population within 50 miles (80 kilometers) of the Site. Based on national averages, medical exposure accounts for 14 percent of the annual dose, and the combined doses from weapons test fallout, consumer and industrial products, and air travel account for approximately 3 percent (Arnett et al. 1993).

4.12.1 Occupational Health and Safety

SRS maintains a network of air monitoring stations on and around the Site to determine the concentrations of radioactive particulates and aerosols in the air (Arnett et al. 1993). Table 4-17 lists average and maximum radionuclide particulate concentrations found in 1992 in air at the F- and H-Areas, SRS boundary, and background [100-mile (160-kilometer) radius] monitoring locations. Table 4-18 lists average and maximum concentrations of tritium in atmospheric moisture during 1992 for the F- and H-Areas, SRS boundary, and background monitoring locations.

Gamma radiation levels measured by thermoluminescent dosimeters in 1992 at the F- and H-Area fences averaged 70 and 74 millirem per year, respectively. Gamma radiation levels, including natural background (terrestrial and cosmic) radiation, measured at the Site perimeter in 1992 yielded an average dose of 35 millirem per year (Arnett et al. 1993).

Soil samples from uncultivated areas provide a measure of the quantity of particulate radioactivity deposited from the atmosphere. Table 4-19 lists maximum measurements of radionuclides in the soil for 1992 at F- and H-Areas, SRS boundary, and background [100-mile (160-kilometer)-radius] monitoring locations. The SRS measured elevated concentrations of plutonium-238 and plutonium-239 around F- and H-Areas, reflecting releases from these areas. From 1955 through 1992, total atmospheric plutonium releases from the F- and H-Areas were approximately 0.7 curie of plutonium-238 and 3 curies of plutonium-239 (Arnett et al. 1992; 1993).

The SRS workers investigated for purposes of assessing occupational radiation exposures belong to the group of involved workers assigned to F- and H-Area facilities. The investigation selected these facilities because they process materials with radiological characteristics similar to the materials being analyzed in this EIS. The dosimetry results for these two involved worker groups are most useful because they depict occupational impacts that are directly relevant to each alternative. The

Table 4-17. Radioactivity in air at the Savannah River Site and vicinity (pCi/m^3).[a]

Location	Gross Alpha	Nonvolatile Beta	SR-89,90[b]	Pu-238[b]	Pu-239[b]
F-Area					
Average	1.80×10^{-3}	1.94×10^{-2}	0.62×10^{-4}	1.26×10^{-5}	8.15×10^{-6}
Maximum	3.55×10^{-3}	5.56×10^{-2}	6.02×10^{-4}	2.64×10^{-5}	2.48×10^{-5}
H-Area					
Average	1.80×10^{-3}	1.93×10^{-2}	2.69×10^{-4}	2.03×10^{-5}	5.14×10^{-6}
Maximum	4.24×10^{-3}	5.39×10^{-2}	2.83×10^{-3}	6.03×10^{-5}	1.41×10^{-5}
Site perimeter					
Average	1.80×10^{-3}	2.30×10^{-2}	0.13×10^{-4}	0.01×10^{-7}	2.40×10^{-7}
Maximum	4.04×10^{-2}	4.95×10^{-2}	4.54×10^{-4}	2.21×10^{-6}	2.76×10^{-6}
Background (100-mile radius)					
Average	1.67×10^{-3}	1.73×10^{-2}	0.49×10^{-4}	0.72×10^{-6}	$<1.00 \times 10^{-6}$
Maximum	3.83×10^{-3}	4.37×10^{-2}	6.89×10^{-4}	1.98×10^{-5}	6.15×10^{-6}

a. Arnett et al. (1993).
b. Monthly composite.

Table 4-18. Tritium in atmospheric moisture at the Savannah River Site (pCi/mL).[a]

Location	Average	Maximum
F-Area	8.67×10^{-5}	2.98×10^{-4}
H-Area	0.99×10^{-3}	6.77×10^{-3}
Site boundary	2.65×10^{-5}	1.03×10^{-4}
Background (100-mile radius)	8.32×10^{-6}	1.08×10^{-5}

a. Arnett et al. (1993).

Table 4-19. Maximum radioactivity concentrations in soil at the Savannah River Site (pCi/g).[a]

Location	Sr-90	Cs-137	Pu-238	Pu-239
F-Area	2.16×10^{-2}	7.19×10^{-1}	4.03×10^{-1}	5.31×10^{-1}
H-Area	2.89×10^{-2}	8.22×10^{-1}	2.13×10^{-2}	5.54×10^{-2}
Site perimeter	(b)	4.84×10^{-1}	2.19×10^{-3}	1.36×10^{-2}
Background (100-mile radius)	1.46×10^{-2}	(b)	2.34×10^{-4}	1.93×10^{-2}

a. Arnett et al. (1992).
b. None detected.

investigation selected two dosimetry periods of record for this analysis: 1983 - 1987 and 1993. The earlier 5-year period included times when materials processing was occurring at a rate that was accelerated in comparison with recent years. The later period includes processing rates that better reflect near-term DOE mission initiatives.

Tables 4-20 and 4-21 list the involved worker dosimetry data for 1983 - 1987 and 1993, respectively. This analysis adapted these data from monitoring data statistics (Matheny 1994a; Matheny 1994b) for operations, maintenance, laboratory, and health protection personnel assigned to the F- and H-Area Canyons and the associated B-Line facilities. The calculated incidences of excess fatal cancer attributable to each facility's collective worker dose are approximately 0.11 and 0.037 for the earlier and later time periods, respectively. Similarly, the highest calculated excess fatal cancer probabilities attributable to average individual worker doses are approximately 0.0003 and 0.0001, respectively. The analysis estimated these health effects using risk coefficients adopted by DOE (DOE 1993).

Table 4-20. Annual involved worker doses, 1983 - 1987.

Facility	Average Worker Dose (rem)	Total Collective Worker Dose (person-rem)
H-Canyon	0.41	36.28
HB-Line	0.49	21.84
F Canyon	0.48	87.25
FB-Line	0.74	124.68
Facilities Average	0.53	—[a]
Facilities Total	—	270.05

a. — = Not applicable.

4.12.2 Public Health and Safety

Table 4-22 summarizes the major sources of exposure for the population within 50 miles (80 kilometers) of the SRS and for the Savannah River water-consuming population in Beaufort and Jasper Counties, South Carolina, and Port Wentworth, Georgia. Most of the sources, such as natural background dose and medical dose, are independent of the presence of the SRS.

Table 4-21. Annual involved worker doses, 1993.

Facility	Average Worker Dose (rem)	Total Collective Worker Dose (person-rem)
H-Canyon	0.17	11.07
HB-Line	0.24	21.97
F Canyon	0.22	9.16
FB-Line	0.24	51.16
Facilities Average	0.22	—[a]
Facilities Total	—	93.36

a. — = Not applicable.

Table 4-22. Major sources of radiation exposure to the public in the vicinity of the Savannah River Site.[a]

Source of Exposure	Dose to average individual (mrem/yr)	Percentage of exposure
Natural background radiation	315	83
Medical radiation	54	14
Consumer and industrial products, fallout, air travel	10	3
Savannah River Site operations	0.22	0.06
Grand Total	380	100

a. Arnett et al. (1993).

Atmospheric releases of radioactive material to the environment from SRS operations from 1990 to 1992 resulted in an average dose of approximately 0.02 millirem per year to individuals in the 50-mile (80-kilometer)-radius population. The collective effective dose equivalent due to atmospheric releases from 1992 SRS operations to the population of 620,100 within 50 miles (80 kilometers) was approximately 6.4 person-rem per year. Atmospheric releases of tritium accounted for more than 90 percent of the offsite population dose; tritium is the only radionuclide of SRS origin that is routinely detected in offsite air (Cummins et al. 1991; Arnett et al. 1992, 1993). Table 4-23 lists average annual atmospheric tritium concentrations in the vicinity of SRS for the three years ending in 1992.

Table 4-23. Average atmospheric tritium concentrations in the vicinity of the Savannah River Site (pCi/m^3).[a]

Location	1992	1991	1990
Onsite	340	250	430
Site perimeter	27	21	32
25-mile radius	11	11	12
100-mile radius	8.3	8.5	8.8

a. Arnett et al. (1993).

From 1990 to 1992, the calculated maximum individual average annual dose from atmospheric releases to a hypothetical individual residing at the SRS boundary was 0.12 millirem (Cummins et al. 1991; Arnett et al. 1992, 1993).

In general, liquid releases of tritium account for more than 99 percent of the total radioactivity introduced into the Savannah River from SRS activities (Arnett et al. 1993). The calculated average annual dose to the maximally exposed individual resulting from liquid releases from 1990 to 1992 was 0.21 millirem (Cummins et al. 1991; Arnett et al. 1992; 1993). From 1990 to 1992 liquid releases of radioactive material to the environment from SRS operations resulted in an average dose of 0.04 millirem per year and 0.05 millirem per year to downstream consumers of drinking water from the Beaufort-Jasper and Port Wentworth water treatment plants, respectively. These doses to the current Beaufort-Jasper river-water-consuming population of about 51,000 and the current Port Wentworth river-water-consuming population of about 20,000 would yield a collective effective dose equivalent to these populations of approximately 3 person-rem per year (Cummins et al. 1991; Arnett et al. 1992, 1993).

The SRS analyzes samples from other environmental media that onsite releases might affect and that might provide a pathway for radiation exposure to the public and Site employees; these include samples of milk, food products, drinking water, wildlife, rainwater, soil, sediment, and vegetation. The 1992 SRS Environmental Report (Arnett et al. 1993) describes the sampling program, monitoring locations, and monitoring results for each of these media.

Major nuclear facilities within 50 miles (80 kilometers) of the SRS include a low-level waste burial site operated by Chem-Nuclear Systems, Inc., near the eastern SRS boundary in Barnwell, South Carolina, and the Georgia Power Company Alvin W. Vogtle Electric Generating Plant, directly across the Savannah River from the SRS. Plant Vogtle began commercial operation in 1987, and its releases are controlled to meet U.S. Nuclear Regulatory Commission requirements.

4.13 Utilities and Energy

This section describes SRS electricity consumption, water consumption, fuel usage, and domestic and industrial wastewater treatment. Table 4-24 contains information on the current status of these items at SRS.

Table 4-24. Current capacities and usage of utilities and energy at SRS.

ELECTRICITY	
Consumption	659,000 megawatt hours per year
Load	75 megavolt-amperes
Peak Demand	130 megavolt-amperes
Capacity	340 megavolt-amperes
WATER	
Groundwater usage	12,490 million liters (3.3 billion gallons) per year
Surface water usage (cooling)	75,700 million liters (20 billion gallons) per year
FUEL	
Oil	28.4 million liters (7.5 million gallons) per year
Coal	210,000 metric tons (230,000 tons) per year
Gasoline	4.7 million liters (1.24 million gallons) per year
WASTEWATER	
Domestic capacity	3.97 million liters (1.05 million gallons) per day
Domestic load	1.89 million liters (0.50 million gallons) per day
Industrial capacity[a,b]	1.64 million liters (433,244 gallons) per day
Industrial load[a]	44,000 liters (11,580 gallons) per day

a. F/H Effluent Treatment Facility only.
b. Design capacity; permitted capacity is about 67 percent of this value.

4.13.1 Electricity

The SRS purchases electric power from the South Carolina Electric and Gas Company (SCE&G) through three purchased power-line interconnects to the SRS transmission grid. The recent total annual power consumption for the SRS was approximately 659,000 megawatt-hours. The average load was 75 megavolt-amperes and the peak demand was about 130 megavolt-amperes. South Carolina

Electric and Gas sources can supply as much as 340 megavolt-amperes to the SRS grid with existing direct connections. The SRS generating station in D-Area can produce an additional 80 megavolt-amperes capacity, although that plant currently produces only process steam. The SRS transmission grid that would provide power to any spent nuclear fuel facilities consists of more than 145 kilometers (90 miles) of 115-kilovolt lines, four switching stations, and 15 substations. Electric service to all major production areas provides parallel redundant capacity to ensure maximum availability and reliability (DOE 1993b).

4.13.2 Water Consumption

Groundwater from a deep confined aquifer supplies domestic and process water for the SRS through approximately 100 production wells. The aquifer system sustains single well yields of about 10.2 million liters (2.7 million gallons) per day. Current usage from this source is about 12.5 billion liters (3.3 billion gallons) per year. The SRS withdraws cooling water for its facilities from the Savannah River at an annual rate of about 75.7 billion liters (20 billion gallons) (DOE 1993b).

4.13.3 Fuel Consumption

Fuels consumed at SRS include oil, coal, and gasoline. SRS facilities and equipment burn approximately 28.4 million liters (7.5 million gallons) of oil each year. This total includes diesel fuel, No. 6 oil, and No. 2 oil. The SRS burns coal and some waste oils in the D-Area powerhouse to produce steam for Site facilities. Current coal usage is about 208,655 metric tons (230,000 tons) per year. SRS vehicles use approximately 4.7 million liters (1.24 million gallons) of gasoline annually. Under the provisions of the Energy Policy Act of 1992, natural gas will replace gasoline on the SRS within the next 10 years. At that time, SRS usage of natural gas would be approximately 12.2 million cubic meters (429 million cubic feet) per year. At present, the SRS consumes no natural gas (DOE 1993b).

4.13.4 Wastewater Treatment

By 1995, the SRS Centralized Sanitary Wastewater Treatment Facility will process most of the domestic effluent on the Site. This centrally located facility has a design capacity of 4 million liters (1.05 million gallons) per day. Once operational, the plant will use about 50 percent of this capacity. In addition, five smaller sanitary treatment plants serve more remote areas of the Site. Facilities for spent nuclear fuel management would use the centralized facility.

The F/H Effluent Treatment Facility (ETF), which decontaminates routine process effluents and accidental radioactive releases from operations, treats industrial wastewater in the F- and H-Areas, where the spent fuel management activities would occur.

Effluent Treatment Facility process operations performed on the waste liquids include neutralization (adjusts pH), submicron filtration (removes suspended solids), activated carbon absorption (removes dissolved organic chemicals), reverse osmosis membrane deionization (removes salts), ion exchange (removes heavy metals), and evaporation (separates radionuclides from aqueous condensate). This facility releases two different streams. The treated water stream is sampled and analyzed to ensure that it meets discharge requirements and then is released to Upper Three Runs Creek via a permitted outfall. The waste concentrat (i.e., bottoms from the evaporator process) is transferred to the H-Area waste tank farm for treatment and disposal in the Z-Area Saltstone facility.

The design capacity for the Effluent Treatment Facility is approximately 600 million liters (158 million gallons) per year. The maximum permitted treatment capacity is about 400 million liters (105.7 million gallons) per year. Under normal operating conditions, the facility treats more than 16,000 cubic meters (26 million gallons) of liquid waste per year (WSRC 1993f).

The influent water load to processes discharging to the permitted outfall includes as much as 205 million liters (54 million gallons) per year of F-Area Canyon process wastewater, 120 million liters (32 million gallons) per year of H-Area Canyon process wastewater, 34 million liters (9 million gallons) per year from the F-Area collection and retention basins, 34 million liters (9 million gallons) per year from the H-Area collection and retention basins, 68 million liters (18 million gallons) per year of Effluent Treatment Facility acid, caustic, flush and rinse water, and similar wastewater from other SRS facilities.

4.14 Materials and Waste Management

The historic national defense mission of the SRS has resulted in the generation of high-level radioactive waste, transuranic waste, low-level radioactive waste (low-activity and intermediate-level), hazardous waste, mixed waste (radioactive and hazardous combined), and sanitary waste (nonhazardous, nonradioactive solid waste). This section discusses the treatment, storage, and disposal of waste at the SRS. Section 4.13 discusses domestic and industrial wastewater treatment.

DOE is preparing an environmental impact statement on Waste Management at the Savannah River Site. The purpose of the EIS is to provide a basis for DOE to select a sitewide strategic approach to managing present and future SRS waste generated as a result of ongoing operations, environmental restoration activities, transition from nuclear production to other missions, and decontamination and decommissioning programs. The Waste Management EIS will support project-level decisions on the operation of specific treatment, storage, and disposal facilities within the near term (10 years or less). In addition, the EIS will provide a baseline for analyses of future waste management activities and a basis for the evaluation of the specific waste management alternatives. The Waste Management EIS will not include management of spent nuclear fuel which is addressed in this document.

DOE treats and stores waste generated from onsite operations in waste management facilities located primarily in E-, F-, H-, N-, S-, and Z-Areas (Figure 4-16). These facilities include the F- and H-Area Effluent Treatment Facility, the High-Level Waste Tank Farms, and the Solid Waste Disposal Facility. The Defense Waste Processing Facility is nearly operational and the Consolidated Incineration Facility is under construction. The SRS places sanitary and inert waste in the Interim Sanitary Landfill and the Burma Road Landfill, respectively.

DOE continues to reduce the amount of waste generated and disposed of at the SRS through waste minimization and treatment programs. DOE accomplishes waste minimization by reducing the volume, toxicity, or mobility of waste before storing or disposing of it. These activities also include more intensive surveying, waste segregation, and use of administrative and engineering controls.

The waste that DOE presently stores on the SRS includes high-level, transuranic, hazardous, mixed waste and some low-level waste. The Site stores high-level waste in underground storage tanks that have received South Carolina Department of Health and Environmental Control industrial wastewater permits, and manages them in accordance with Clean Water Act, Resource Conservation and Recovery Act, and DOE requirements. The SRS stores transuranic mixed waste on interim-status storage pads in accordance with South Carolina Department of Health and Environmental Control requirements and DOE Orders. Hazardous and mixed waste is placed in permitted or interim-status

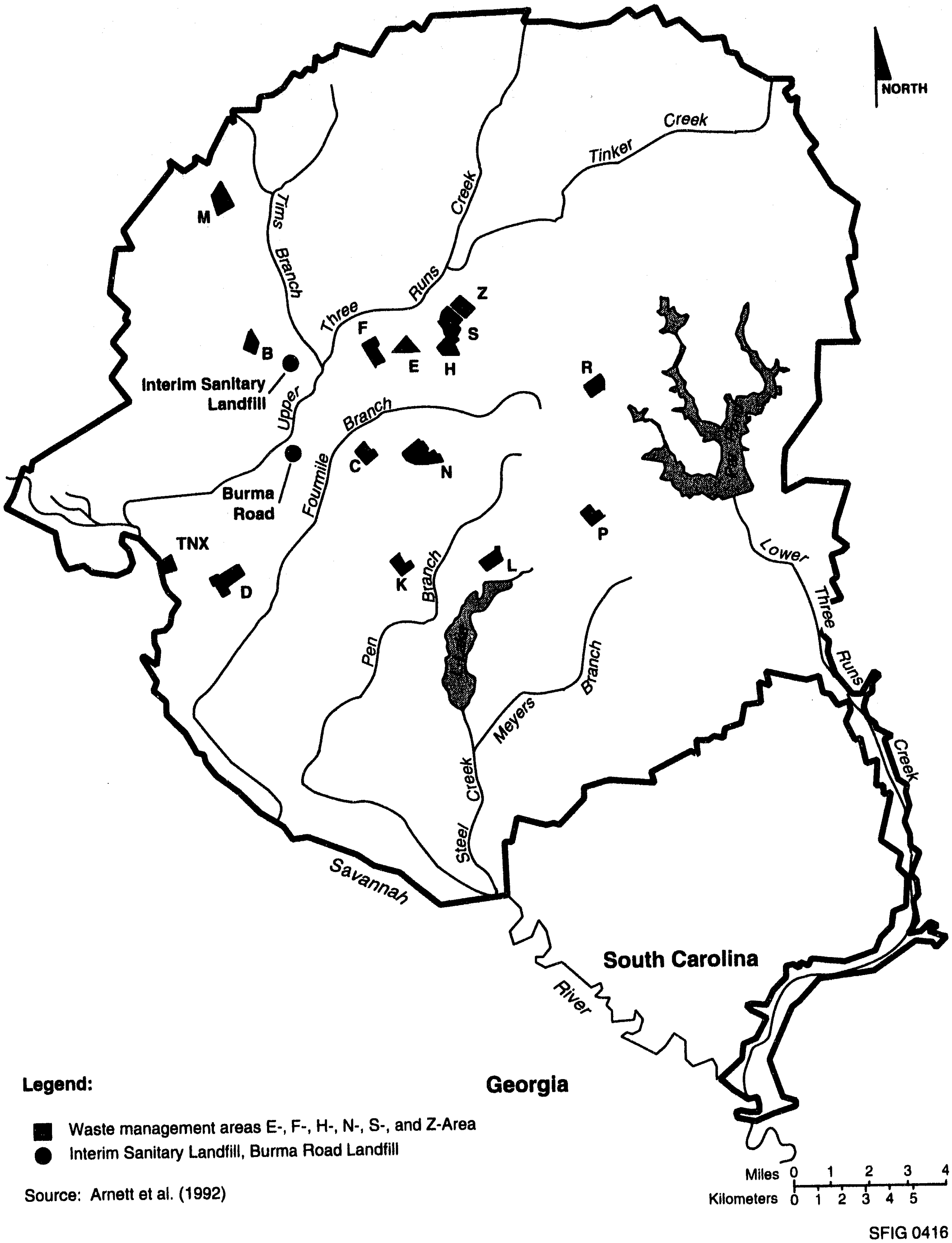


Figure 4-16. Waste management facilities at the Savannah River Site.

storage in the Hazardous Waste Storage Facilities (both buildings and pads) and in the mixed waste storage buildings.

Figure 4-17 shows the high-level liquid waste management process at the SRS. Figure 4-18 shows the process for handling all other forms of solid waste at the Site.

Table 4-25 is a forecast of annual waste generation for all waste forms except sanitary and high-level waste (WSRC 1994d). The volumes listed do not include waste related to decontamination and decommissioning because DOE has not yet completed the planning of these activities. Section 5.14 discusses potential consequences of spent nuclear fuel activities as they relate to the alternative interim storage and treatment scenarios.

4.14.1 High-Level Waste

The SRS generated high-level waste from the recovery of nuclear materials from spent fuel and target processing in the F- and H-Areas. It is stored in 50 underground tanks. These tanks also store other radioactive waste effluents (primarily low-level radioactive waste such as aqueous process waste, including purge water from storage basins for irradiated reactor fuel or fuel elements). The high-level waste is segregated by heat generation rate, neutralized to excess alkalinity for storage tank corrosion protection, and stored to permit the decay of short-lived radionuclides before its volume is reduced by evaporation. Evaporators concentrate alkaline waste to reduce original volumes and to immobilize it as crystallized salt by successive evaporations of the liquid supernate. The SRS treats the evaporator overheads in cesium removal columns before transferring them to the F- and H-Area Effluent Treatment Facility. [DOE is preparing an EIS for the Interim Management of Nuclear Materials at the SRS (59 FR 12588, 3/17/94)]. The SRS processes the sludge and salt to prepare them for vitrification at the Defense Waste Processing Facility (high-level waste), when it becomes operational, or stabilization at the Z-Area Saltstone Facility (low-level waste). DOE is preparing a Supplemental EIS related to Defense Waste Processing Facility operations (59 FR 16499, 4/6/94).

By December 31, 1991, DOE had stored approximately 127.9 million liters (33.8 million gallons) of high-level radioactive waste on the Site. Estimates of current tank capacity and high-level waste forecasts should be available in 1995. In general, however, due to a number of factors, the most important of which has been the extended outage of the evaporators, the estimated inventory of waste in the high-level tanks is greater than 90 percent of existing capacity (WSRC 1994b). DOE is

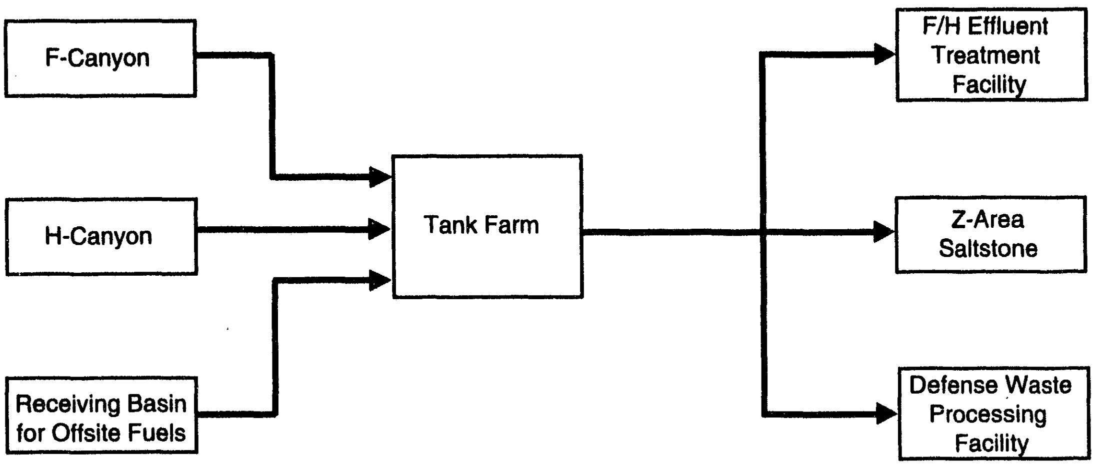
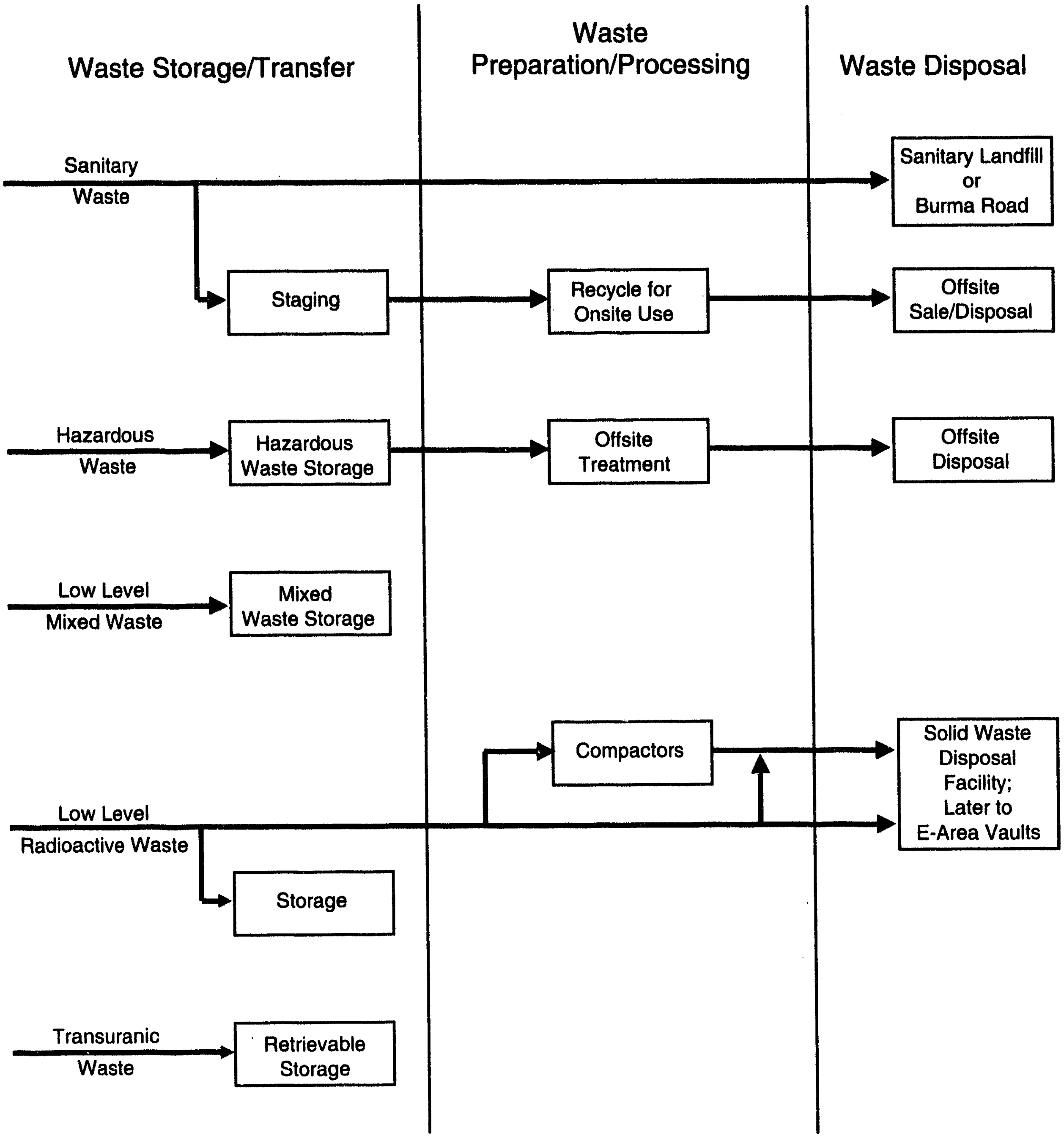


Figure 4-17. Flow diagram for high-level radioactive waste handling at the Savannah River Site.



Source: WSRC (1994c)

SFIG 0418

Figure 4-18. Flow diagram for waste handling at the Savannah River Site.

Table 4-25. Average annual waste generation forecast for Savannah River Site (cubic meters).[a,b]

Waste Type	FY94	FY95	FY96
Transuranic	670	860	760
Low-Level			
Low-Activity	21,350	17,680	17,970
Intermediate-Level	940	580	740
Hazardous	140	130	100
Mixed	120	130	110

a. Source: WSRC (1994d).
b. To convert cubic meters to cubic feet, multiply by 35.314.

constructing a replacement high-level waste tank evaporator to augment or replace existing evaporators.

4.14.2 Transuranic Waste

At present, DOE uses three methods of retrievable storage for transuranic waste at SRS, based on the time of generation. Transuranic waste generated before 1974 is buried in approximately 120 belowgrade concrete culverts in the Solid Waste Disposal Facility. Transuranic waste generated from 1974 to 1985 is stored on five concrete pads and one asphalt pad that have been covered with approximately 1.2 meters (4 feet) of native soil. DOE stores waste generated since 1985 on 13 additional concrete pads that are not covered with soil. Pads 1 through 17 operate under Interim Status approved by the South Carolina Department of Health and Environmental Control. DOE uses Pads 18 through 19, which are not required to have interim status, to manage nonhazardous transuranic wastes only.

The SRS stores wastes containing 10 to 100 nanocuries per gram of transuranic material with transuranic waste until it can complete Site-specific radiological performance assessments, which will provide disposal limits for transuranic isotopes. SRS transuranic waste inventories and forecasts include both transuranic waste and the 10- to 100-nanocuries-per-gram transuranic wastes.

At the end of 1993, the SRS had approximately 9,900 cubic meters (350,000 cubic feet) of transuranic waste in storage (WSRC 1994c). Based on the 1994-to-1996 average annual generation rate forecast, the Site generates approximately 760 cubic meters (27,000 cubic feet) of transuranic waste annually. Transuranic mixed waste (transuranic and hazardous combined) accounts for approximately 110 cubic meters (3,900 cubic feet) of this volume (WSRC 1994d). DOE is evaluating available storage space for transuranic mixed waste to mitigate any storage capacity deficit.

4.14.3 Mixed Low-Level Waste

The SRS mixed waste program consists primarily of providing safe storage until treatment and disposal facilities are available. The current volume of mixed low-level waste at the SRS is 1,700 cubic meters (60,000 cubic feet) (WSRC 1994c). Based on the 1994-to-1996 average annual generation forecast, the Site generates approximately 118 cubic meters (4,170 cubic feet) of mixed low-level waste annually (WSRC 1994d). DOE is evaluating available storage space to determine when the SRS will exceed its capacity. However, DOE is constructing a Consolidated Incineration Facility in H-Area, which will treat mixed, hazardous, and low-level waste. When the incinerator is operational, it will treat approximately 90 percent of the existing mixed-waste inventory, and storage capacity will expand (WSRC 1993f).

4.14.4 Low-Level Waste

The SRS packages low-level waste for disposal on the Site in accordance with the waste category and its estimated surface dose rate. The Site places low-activity waste in carbon steel boxes and deposits it in an Engineered Low-Level Trench (ELLT). The trenches are several acres in size by 6 meters (20 feet) deep and have sloped sides and floor, allowing drainage to a collection sump. When the trenches are full, DOE backfills and covers them with at least 1.8 meters (6 feet) of soil. The Site packages intermediate-level wastes according to the waste form and disposes of them in slit trenches. DOE will store long-lived wastes, such as resins, until the Long-Lived Waste Storage Building, currently under construction, becomes operational. This building will provide storage until DOE develops treatment and disposal technologies.

The SRS is developing a new disposal facility, known as the E-Area Vault (EAV). This facility will include vaults for low-activity waste, intermediate-level non-tritium waste, and intermediate-level tritium waste.

Based on the 1994-to-1996 average annual generation forecast, the Site generates approximately 19,000 cubic meters (671,400 cubic feet) of low-activity waste and 750 cubic meters (26,600 cubic feet) of intermediate-level waste annually. DOE expects that the Consolidated Incineration Facility will begin operations by the second quarter of Fiscal Year 1996; this facility will have the capability of annually processing as much as 15,850 cubic meters (560,000 cubic feet) of boxed low-activity waste and approximately 186 cubic meters (6,600 cubic feet) of hazardous and mixed waste.

4.14.5 Hazardous Waste

DOE stores hazardous wastes generated at various SRS facilities in buildings in the B- and N-Areas, and on the Solid Waste Storage Pads. The Resource Conservation and Recovery Act regulates these wastes.

The inventory of hazardous waste in storage at the SRS is about 1.6 million kilograms (3.6 million pounds), occupying a volume of about 2,430 cubic meters (86,000 cubic feet) (WSRC 1994c). Based on the 1994-to-1996 average annual generation rate forecast, the Site generates approximately 124 cubic meters (4,370 cubic feet) of hazardous waste annually (WSRC 1994d).

4.14.6 Sanitary Waste

The SRS disposes of most of its solid sanitary waste in onsite landfills, the most recent of which began operation in 1985. Current disposal operations include the Interim Sanitary Landfill. About 30 trucks per work day arrive at this facility carrying approximately 18,125 kilograms (40,000 pounds) of waste that, after compaction, occupies approximately 115 cubic meters (150 cubic yards) of landfill space. The recent implementation of SRS paper and aluminum can recycling programs and disposal of office waste off the Site in a commercial landfill has increased the projected life of the landfill to the fourth quarter of 1996 (WSRC 1994c).

DOE also maintains an inert material landfill on the Site near Burma Road. This facility receives demolition and construction debris. DOE is evaluating the construction of a new SRS sanitary landfill or the use of a commercial landfill.

4.14.7 Hazardous Materials

The SRS 1993 Tier II emergency and hazardous chemical inventory lists 205 reportable hazardous substances present on the Site in excess of the 10,000-pound (4,536-kilogram) threshold quantity (WSRC 1994e). The number and the total weight of any hazardous chemicals used on the Site change daily in response to use. The annual Superfund Amendments and Reauthorization Act

(SARA) reports for the SRS include listings of hazardous materials used or stored on the Site during each year.

5. ENVIRONMENTAL CONSEQUENCES

5.1 Overview

This chapter discusses the potential environmental consequences for each spent nuclear fuel management alternative described in Chapter 3. The representative host site locations, as described in Chapter 2, are the F- and H-Areas and an undeveloped site close to H-Area. These sites are representative of available areas that could support spent fuel management missions. Based on generic facility characteristics, this chapter analyzes representative consequences in terms of the environmental attributes of the potential host areas and the Savannah River Site (SRS) at large, as described in Chapter 4. Table 3-2 compares the environmental consequences of each alternative. The impacts associated with the construction and operation of a Navy Expended Core Facility are not included in this chapter, but are included in Appendix D of this Environmental Impact Statement.

5.2 Land Use

Overall environmental impacts on land use by any of the alternatives would be small because the U.S. Department of Energy (DOE) would construct most new facilities in F- and H-Areas, which are already dedicated to industrial use and which previous activities have disturbed. New construction on the undeveloped representative host site near H-Area would probably be necessary only for the construction of a dry storage vault.

The Centralization Alternative (Alternative 5), under which DOE would transfer all spent nuclear fuel to the SRS, would result in the greatest changes in land use. Under this alternative, the SRS would dedicate between 70 and 100 acres (0.3 and 0.4 square kilometer) for use in spent nuclear fuel management; the exact location and size of the area affected would depend on whether DOE chose to use the wet storage, dry storage, or processing option. Of this affected area, a maximum of approximately 100 acres (0.4 square kilometer) would change from managed pine forest to industrial use. The remaining area would retain its current use as an industrial facility, with DOE performing some modifications or new construction on already disturbed areas.

DOE would retain under its control any lands supporting the spent nuclear fuel management program for the life of the project. No alternative would require the acquisition of public lands.

5.3 Socioeconomics

Socioeconomic consequences resulting from the implementation of any of the alternatives would relate primarily to changes in employment within the region of influence (ROI). DOE has based the analysis in the following section on estimated employment and population data for each SRS spent nuclear fuel alternative, as listed in Table 5-1. The population within the region of influence in 1995 is estimated to be approximately 462,000. The labor force will be about 257,000 persons of which about 242,000 will be employed.

DOE expects the employment level at the Site to decline from about 20,000 (in 1995) to about 18,700 (in 2004) as the SRS mission is redefined. This anticipated decline would be somewhat offset by the jobs created by the spent nuclear fuel management activities. Therefore, none of the alternatives would require additional operations employees because the SRS could fill all operational positions through the reassignment of existing workers. Consequently, this analysis addresses only employment impacts from construction activities. Given the natural variation in construction employment levels, the analysis could not accurately determine the reassignment of existing construction workers. As a result, this assessment analyzed the maximum potential impact, which assumes that all construction employment would represent new jobs that in-migrating workers would fill.

DOE estimated total employment impacts using the Regional Input-Output Modeling System that the U.S. Bureau of Economic Analysis developed for the SRS region of influence. This assessment also analyzed changes in population based on historic data that indicate that 90 percent of SRS employees live in the six-county region.

5.3.1 Potential Impacts

Table 5-1 lists direct increases in construction employment for each alternative and the corresponding change in population. As listed, potential impacts to socioeconomic resources would be smallest under Alternative 1 (No Action) and would be greatest under Option 5b (Centralization - Wet Storage). Therefore, Option 5b provides the bounding case for maximum potential impacts to socioeconomic resources.

Table 5-1. Direct construction employment and total population changes by alternative, 1995-2004.

Alternative	1995[a]	1996[a]	1997[a]	1998[a]	1999[a]	2000	2001	2002	2003	2004
Alternative 1-										
Employment[a]	50	50	50	50	50	50	50	50	50	50
Population	200	150	150	100	100	100	100	100	100	100
Option 2a-										
Employment	50	50	50	50	50	200	400	600	500	200
Population	200	150	150	100	100	850	1,550	2,250	2,000	750
Option 2b-										
Employment	50	50	50	50	50	200	400	600	500	200
Population	100	150	150	100	100	850	1,550	2,250	2,000	750
Option 2c-										
Employment	50	50	50	50	50	200	350	550	500	150
Population	200	150	150	100	100	700	1,350	2,050	1,850	600
Option 3a-										
Employment	50	50	50	50	50	200	400	600	500	200
Population	200	150	150	100	100	850	1,550	2,250	2,000	750
Option 3b-										
Employment	50	50	50	50	50	200	400	650	600	250
Population	200	150	150	100	100	800	1,600	2,550	2,400	900
Option 3c-										
Employment	50	50	50	50	50	200	350	550	500	150
Population	200	150	150	100	100	700	1,350	2,050	1,850	600
Option 4a-										
Employment	50	50	50	50	50	200	400	650	600	250
Population	200	150	150	100	100	800	1,600	2,550	2,400	900
Option 4b-										
Employment	50	50	50	50	50	200	400	650	600	250
Population	200	150	150	100	100	800	1,600	2,550	2,400	900
Option 4c-										
Employment	50	50	50	50	50	200	350	550	500	150
Population	200	150	150	100	100	700	1,350	2,050	1,850	600
Option 4d-										
Employment	50	50	50	50	50	300	500	700	650	250
Population	200	200	150	150	150	1,100	1,900	2,800	2,500	900
Option 4e-										
Employment	50	50	50	50	50	250	500	800	800	300
Population	200	200	150	150	150	1,000	2,000	3,200	3,000	1,100
Option 4f-										
Employment	50	50	50	50	50	200	450	650	600	200
Population	200	200	150	150	150	850	1,700	2,550	2,350	700
Option 4g-										
Employment	50	50	50	50	50	100	150	200	100	100
Population	200	150	150	100	100	250	500	700	450	300

Table 5-1. (continued).

Alternative	1995[a]	1996[a]	1997[a]	1998[a]	1999[a]	2000	2001	2002	2003	2004
Option 5a-										
Employment	50	50	50	50	50	900	1,750	2,550	2,500	2,450
Population	200	150	150	100	100	3,500	6,800	9,900	9,700	9,450
Option 5b-										
Employment	50	50	50	50	50	1,000	1,900	2,700	2,650	2,600
Population	200	150	150	100	100	3,850	7,450	10,550	10,350	10,100
Option 5c-										
Employment	50	50	50	50	50	900	1,750	2,550	2,500	2,450
Population	200	150	150	100	100	3,500	6,800	9,900	9,700	9,500
Option 5d-										
Employment	50	50	50	50	50	100	150	200	100	100
Population	200	150	150	100	100	250	500	700	450	300

a. Construction is related to renovation of reactor basin and Receiving Basin for Offsite Fuels.

Table 5-2 lists indirect employment and corresponding population changes associated with construction phase activities under Option 5b. As listed, the number of full-time construction workers required to support the implementation of this option from 1995 to 2004 would range from approximately 50 to 2,700. When added to the indirect employment of 1,600 jobs in the peak year (2002), the total employment impact in the region would be approximately 4,300 employees.

Table 5-2. Estimated increases in employment and population related to construction activities for Option 5b, from 1995 to 2004. ROI refers to the six-county region of influence.

Factor	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004
Direct employment	50	50	50	50	50	1,000	1,900	2,700	2,650	2,600
Secondary employment	30	30	30	30	30	600	1,100	1,600	1,550	1,500
Total employment change	80	80	80	80	80	1,600	3,000	4,300	4,200	4,100
% Change in ROI labor force	0.03	0.03	0.03	0.03	0.03	0.54	1.00	1.41	1.36	1.32
% Change in ROI employment	0.03	0.03	0.03	0.03	0.03	0.57	1.06	1.50	1.45	1.40
Population change (in region)	200	150	150	100	100	3,850	7,450	10,550	10,350	10,100
% Change in ROI population	0.04	0.03	0.03	0.02	0.02	0.81	1.56	2.21	2.16	2.11

Assuming in-migrating workers filled all jobs, the regional labor force and employment would increase by 1.4 percent and 1.5 percent, respectively. These changes would be temporary and would have no adverse impact on the region. After 2004, employment would gradually decline to a relatively constant level of about 50 jobs.

Based on historic data, approximately 90 percent of new employees would live within the six-county region of influence. Assuming each new employee represented one household with 2.72 persons per household, there would be approximately 10,550 additional people in the region during the peak year (2002). These changes would be temporary and would represent an estimated 2.2-percent increase in baseline population levels. Given this minor change in population, DOE expects potential impacts on the demand for community resources and services such as housing, schools, police, health care, and fire protection to be negligible.

Because all the other alternatives would require fewer employees, they would result in smaller changes than those listed in Table 5-2, and would have no adverse impacts on socioeconomic resources in the region of influence.

5.4 Cultural Resources

A Programmatic Memorandum of Agreement (SRARP 1989) between the DOE Savannah River Operations Office, the South Carolina State Historic Preservation Office, and the Advisory Council on Historic Preservation, ratified on August 24, 1990, is the instrument for the management of cultural resources at the SRS. DOE uses this memorandum to identify cultural resources, assess them in terms of eligibility for the National Register of Historic Places, and develop mitigation plans for affected resources in consultation with the State Historic Preservation Officer. DOE would comply with the terms of the memorandum for all activities needed to support spent nuclear fuel management actions.

The potential for adverse impacts on cultural resources would be smallest under Alternative 1 (No Action) and would be greatest under Alternative 5 (Centralization). Any facilities that DOE would construct in F- and H-Areas, north of Road E (Alternatives 1-5), would be in Sensitivity Zones 2 and 3. Section 4.4 describes these zones. The undeveloped representative host site south and east of H-Area (Alternative 5) is in Sensitivity Zone 3. Although there are no known archeological sites in the area, it has never been surveyed. Surveying being conducted near F-Area (north of Road C and west of Road 4 along Upper Three Runs Creek) has recorded some historic and prehistoric sites. However, DOE expects no impacts in F- and H-Areas due to their extensive industrial development. Until DOE has determined the precise locations of facilities connected with any of the alternatives, it cannot predict impacts on cultural resources in the undeveloped site area (Sassaman 1993, 1994). However, DOE would mitigate, through avoidance or removal, impacts to potentially significant resources that future site surveys might discover.

5.5 Aesthetic and Scenic Resources

None of the alternatives for spent nuclear fuel management at the SRS would have adverse consequences on scenic resources or aesthetics. Most new construction would be in F- or H-Area, both of which are already dedicated to industrial use. New construction on the undeveloped site, which would occur primarily under Alternative 5, would be adjacent to H-Area in an already heavily industrialized portion of the SRS. In all cases, new construction would not be visible off the Site or from public access roads on the Site. No alternative would produce emissions to the atmosphere that would be visible or would indirectly reduce visibility.

5.6 Geologic Resources

The SRS contains no unique geologic features or minerals of economic value. Therefore, DOE anticipates no impacts to geologic resources at the SRS from any of the spent nuclear fuel management alternatives.

Other sections in this chapter consider the relationships of the Site's specific geology and the region's historic and analyzed seismicity to the local environment and to SRS spent nuclear fuel-related structures and facilities. Section 5.8 discusses the consequences of analyzed seismic events on both surface-water and groundwater resources. Section 5.15 describes estimates of risk that consider both the probability of and the consequences from a wide range of seismic events, ranging from local and regional historically documented earthquakes to postulated lower probability, higher consequence events.

The accident analyses in this chapter, which DOE based on information from approved safety analysis reports for applicable facilities, address the frequency and consequences of historic earthquakes, as well as postulated less likely, but more damaging, seismic events. DOE has evaluated the consequences from seismic challenges to the facilities and structures up to 0.2*g* lateral ground acceleration.

5.7 Air Quality Consequences

The SRS is in compliance with both Federal and state ambient air quality standards for criteria and toxic air pollutants. As shown in the following tables, the predicted incremental air pollutant impacts would not contribute to exceeding either the National Ambient Air Quality Standards or South Carolina's Ambient Air Quality Standards.

DOE performed analyses using computer models in order to assess the potential air quality impacts of operations under each of the spent nuclear fuel management alternatives. This section describes the results of these analyses. All the concentrations discussed below are ground-level estimations based on results from the ISC2 and FDM models for nonradiological pollutants, and MAXIGASP- and POPGASP SRS-climatology-specific models for radionuclides. The analyses assume that facility operations would result in both radiological and nonradiological emissions. DOE assessed construction impacts qualitatively in relation to the land area to be disturbed under each alternative.

Nonradiological Emissions. DOE analyzed the potential incremental impacts of only those substances for which it expects releases to the atmosphere during the normal operation of spent nuclear fuel facilities. The nonradiological releases evaluated for each alternative include seven criteria pollutants and 23 toxic pollutants. DOE selected the toxic substances for analysis by comparing the anticipated chemical usage at the proposed spent nuclear fuel facilities to the list of 257 toxic air pollutants in the South Carolina Air Pollution Regulations (R.61-62.5, Standard 8). The SRS modeled potential emissions of the listed toxic chemicals that DOE anticipates would be used during spent nuclear fuel activities. The following subsections discuss the results for both criteria and toxic pollutants. Tables 5-3 and 5-4 list the estimated maximum incremental concentrations of these pollutants at the Site boundary, while Tables 5-5 and 5-6 contain the incremental rates of release.

Table 5-3. Estimated incremental air quality impacts at the Savannah River Site boundary from operations of spent nuclear fuel alternatives - criteria pollutants ($\mu g/m^3$).[a]

Pollutant[b]	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[c]	Incremental Concentrations from Alternatives: No Action 1	Decentralization 2a	Decentralization 2b	Decentralization 2c	1992/1993 Planning Basis 3a	1992/1993 Planning Basis 3b	1992/1993 Planning Basis 3c
CRITERIA POLLUTANTS ($\mu g/m^3$)											
Carbon monoxide	8-hour	10,000	818	23	<0.01	0.1	0.1	4.3	0.1	0.1	4.3
	1-hour	40,000	3,553	180	<0.01	0.8	0.8	32	0.8	0.8	32
Ozone (as VOC)	1-hour	245	N/A[d]	N/A[d]	1.6	0.3	0.3	2.6	0.3	0.3	2.6
Nitrogen oxides	Annual geometric mean	100	30	4	<0.01	0.01	<0.01	11.00	<0.01	<0.01	11.0
Particulate matter (<10μm)	Annual	50	9	3	—	—	—	<0.01	—	—	0.01
	24-hour	150	93	56	—	—	—	0.40	—	—	0.40
Total suspended particulates (TSP)	Annual	75	20	11	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Sulfur dioxide	Annual	80	18	10	—	<0.01	<0.01	0.01	<0.01	<0.01	0.01
	24-hour	365	356	185	—	0.01	0.01	0.43	0.01	0.01	0.43
	3-hour	1,300	1,210	634	—	0.05	0.05	3.2	0.05	0.05	3.2
Lead	Calendar quarter mean	1.5	<0.01	<0.01	—	—	—	—	—	—	—
Gaseous Fluorides (as HF)	1-month	0.8	0.11	0.03	—	—	—	0.02	—	—	0.02
	1-week	1.6	0.6	0.15	—	—	—	0.10	—	—	0.10
	24-hour	2.9	1.20	0.31	—	—	—	0.20	—	—	0.20
	12-hour	3.7	2.40	0.62	—	—	—	0.40	—	—	0.40

Table 5-3. (continued).

Pollutant[b]	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[c]	Incremental Concentrations from Alternatives: Regionalization A: 4a	4b	4c	Regionalization B: 4d	4e	4f	4g
CRITERIA POLLUTANTS ($\mu g/m^3$)											
Carbon monoxide	8-hour	10,000	818	23	0.2	0.2	4.3	0.2	0.2	5.5	—
	1-hour	40,000	3,553	180	1.2	1.2	32	1.5	1.5	41	—
Ozone (as VOC)	1-hour	245	N/A[d]	N/A[d]	0.5	0.5	2.6	0.6	0.6	3.3	1.4
Nitrogen oxides	Annual geometric mean	100	30	4	<0.01	<0.01	11	<0.01	<0.01	14	—
Particulate matter (<10μm)	Annual	50	9	3	—	—	0.01	—	—	0.01	—
	24-hour	150	93	56	—	—	0.4	—	—	0.5	—
Total suspended particulates (TSP)	Annual	75	20	11	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	—
Sulfur dioxide	Annual	80	18	10	<0.01	<0.01	0.01	<0.01	<0.01	0.01	—
	24-hour	365	356	185	0.02	0.02	0.43	0.02	0.02	0.55	—
	3-hour	1,300	1,210	634	0.09	0.09	3.2	0.11	0.11	4.1	—
Lead	Calendar quarter mean	1.5	<0.01	<0.01	—	—	—	—	—	—	—
Gaseous Fluorides (as HF)	1-month	0.8	0.11	0.03	—	—	0.02	—	—	0.02	—
	1-week	1.6	0.6	0.15	—	—	0.10	—	—	0.13	—
	24-hour	2.9	1.20	0.31	—	—	0.20	—	—	0.25	—
	12-hour	3.7	2.40	0.62	—	—	0.40	—	—	0.51	—

Table 5-3. (continued).

	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[e]	Incremental Concentrations from Alternatives: Centralization 5a	5b	5c	5d
CRITERIA POLLUTANTS (µg/m³)								
Carbon monoxide	8-hour	10,000	818	23	1.0	1.0	5.1	—
	1-hour	40,000	3,553	180	6.7	6.7	37	—
Ozone (as VOC)	1-hour	245	N/A[d]	N/A[d]	1.4	1.4	3.1	1.4
Nitrogen oxides	Annual geometric mean	100	30	4	0.04	0.04	11.1	—
Particulate matter (<10µm)	Annual	50	9	3	—	—	0.01	—
	24-hour	150	93	56	—	—	0.40	—
Total suspended particulates (TSP)	Annual	75	20	11	<0.01	<0.01	<0.01	—
Sulfur dioxide	Annual	80	18	10	<0.01	<0.01	0.02	—
	24-hour	365	356	185	0.09	0.09	0.49	—
	3-hour	1,300	1,210	634	0.50	0.50	3.5	—
Lead	Calendar quarter mean	1.5	<0.01	<0.01	—	—	—	—
Gaseous Fluorides (as HF)	1-month	0.8	0.11	0.03	—	—	0.02	—
	1-week	1.6	0.6	0.15	—	—	0.10	—
	24-hour	2.9	1.20	0.31	—	—	0.10	—
	12-hour	3.7	2.40	0.62	—	—	0.40	—

— = No impact.

a. Maximum modeled ground-level concentration at SRS perimeter unless higher offsite concentrations are otherwise specified.

b. Major pollutants of concern regarding spent nuclear fuel management activities.

c. Most stringent Federal and state regulatory standards [40 CFR Part 50 "National Ambient Air Quality Standards," SCDHEC R.61-62.5, Standard 2, "Ambient Air Quality Standards," and SCDHEC R.61-62.5, Standard 8, "Toxic Air Pollutants"].

d. Measurement data currently unavailable.

e. Maximum operational air pollutant emissions projected for baseline year 1995. Concentration estimates based on actual emissions from all SRS sources for calendar year 1990 plus maximum potential emissions for sources permitted through December 1992.

Table 5-4. Estimated incremental air quality impacts at the Savannah River Site boundary from operations of spent nuclear fuel alternatives - toxic pollutants ($\mu g/m^3$).[a]

					Incremental Concentrations from Alternatives						
					No Action	Decentralization			1992/1993 Planning Basis		
Pollutant[b]	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[d]	1	2a	2b	2c	3a	3b	3c
TOXIC POLLUTANTS ($\mu g/m^3$)											
Nitric acid	24-hour	125	51	6.7	—	—	—	<0.01	—	—	<0.01
1,1,1,- Trichloroethane	24-hour	9,550	81	22	<0.01	<0.01	<0.01	0.01	<0.01	<0.01	0.01
Benzene	24-hour	150	32	31	—	—	—	0.04	—	—	0.04
Ethanolamine	24-hour	200	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	0.58	0.12	—	—	—	<0.01	—	—	<0.01
Ethylene glycol	24-hour	650	0.20	0.08	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	7.5	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronapthalene	24-hour	1.0	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	200	0.21	0.07	<0.01	<0.01	<0.01	0.04	<0.01	<0.01	0.04
Manganese	24-hour	25	0.82	0.10	—	—	—	<0.01	—	—	<0.01
Methyl alcohol	24-hour	1,310	2.9	0.51	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	6.0	0.99	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	3.0	0.51	—	—	—	<0.01	—	—	<0.01
Methylene chloride	24-hour	515	10.5	1.8	—	—	—	0.02	—	—	0.02
Naphthalene	24-hour	1,250	0.01	0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	0.03	0.03	—	—	—	<0.01	—	—	<0.01
Phosphorus	24-hour	0.5	<0.001	<0.001	—	—	—	<0.001	—	—	<0.001
Sodium hydroxide	24-hour	20	0.01	0.01	—	—	—	<0.01	—	—	<0.01
Toluene	24-hour	2,000	9.3	1.6	<0.01	<0.01	<0.01	0.04	<0.01	<0.01	0.04
Trichloroethylene	24-hour	6,750	4.8	1.0	—	—	—	<0.01	—	—	<0.01
Vinyl acetate	24-hour	176	0.06	0.02	—	—	—	<0.01	—	—	<0.01
Xylene	24-hour	4,350	39	3.8	0.01	0.01	0.01	0.05	0.01	0.01	0.05

Table 5-4. (continued).

Pollutant[b]	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[d]	Incremental Concentrations from Alternatives						
					Regionalization A			Regionalization B			
					4a	4b	4c	4d	4e	4f	4g
TOXIC POLLUTANTS ($\mu g/m^3$)											
Nitric acid	24-hour	125	51	6.7	—	—	1.0	—	—	1.3	—
1,1,1,- Trichloroethane	24-hour	9,550	81	22	<0.01	<0.01	0.01	<0.01	<0.01	0.01	<0.01
Benzene	24-hour	150	32	31	—	—	0.04	—	—	0.05	—
Ethanolamine	24-hour	200	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	0.58	0.12	—	—	<0.01	—	—	<0.01	—
Ethylene glycol	24-hour	650	0.20	0.08	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	7.5	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronapthalene	24-hour	1.0	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	200	0.21	0.07	<0.01	<0.01	0.04	<0.01	<0.01	0.05	<0.01
Manganese	24-hour	25	0.82	0.10	—	—	<0.01	—	—	<0.01	—
Methyl alcohol	24-hour	1,310	2.9	0.51	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	6.0	0.99	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	3.0	0.51	—	—	<0.01	—	—	<0.01	—
Methylene chloride	24-hour	515	10.5	1.8	—	—	0.02	—	—	0.02	—
Naphthalene	24-hour	1,250	0.01	0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	0.03	0.03	—	—	<0.01	—	—	<0.01	—
Phosphorus	24-hour	0.5	<0.001	<0.001	—	—	<0.001	—	—	<0.001	—
Sodium hydroxide	24-hour	20	0.01	0.01	—	—	<0.01	—	—	<0.01	—
Toluene	24-hour	2,000	9.3	1.6	<0.01	<0.01	0.04	<0.01	<0.01	<0.05	<0.01
Trichloroethylene	24-hour	6,750	4.8	1.0	—	—	<0.01	—	—	<0.01	—
Vinyl acetate	24-hour	176	0.06	0.02	—	—	<0.01	—	—	<0.01	—
Xylene	24-hour	4,350	39	3.8	0.01	0.01	0.05	0.01	0.01	0.06	0.01

Table 5-4. (continued).

Pollutant[b]	Averaging Time	Regulatory Standard[c]	Maximum Potential Concentration	Actual Concentration[d]	Incremental Concentrations from Alternatives: Centralization 5a	5b	5c	5d
TOXIC POLLUTANTS ($\mu g/m^3$)								
Nitric acid	24-hour	125	51	6.7	—	—	1.0	—
1,1,1,- Trichloroethane	24-hour	9,550	81	22	<0.01	<0.01	0.01	<0.01
Benzene	24-hour	150	32	31	—	—	0.04	—
Ethanolamine	24-hour	200	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	0.58	0.12	—	—	<0.01	—
Ethylene glycol	24-hour	650	0.20	0.08	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	7.5	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronapthalene	24-hour	1.0	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	200	0.21	0.07	<0.01	<0.01	0.04	<0.01
Manganese	24-hour	25	0.82	0.10	—	—	<0.01	—
Methyl alcohol	24-hour	1,310	2.9	0.51	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	6.0	0.99	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	3.0	0.51	—	—	<0.01	—
Methylene chloride	24-hour	515	10.5	1.8	—	—	0.02	—
Naphthalene	24-hour	1,250	0.01	0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	0.03	0.03	—	—	<0.01	—
Phosphorus	24-hour	0.5	<0.001	<0.001	—	—	<0.001	—
Sodium hydroxide	24-hour	20	0.01	0.01	—	—	<0.01	—
Toluene	24-hour	2,000	9.3	1.6	<0.01	<0.01	0.04	<0.01
Trichloroethylene	24-hour	6,750	4.8	1.0	—	—	<0.01	—
Vinyl acetate	24-hour	176	0.06	0.02	—	—	<0.01	—
Xylene	24-hour	4,350	39	3.8	0.01	0.01	0.05	0.01

— No impact.
\+ Not available.
a. Maximum modeled ground-level concentration at SRS perimeter unless higher offsite concentrations are otherwise specified.
b. Major pollutants of concern regarding spent nuclear fuel.
c. Most stringent Federal and state regulatory standards [40 CFR 50 "National Ambient Air Quality Standards," SCDHEC R.61-62.5, Standard 2, "Ambient Air Quality Standards," and SCDHEC R.61-62.5, Standard 8, "Toxic Air Pollutants"].
d. Maximum operational air pollutant emissions projected for baseline year 1995. Concentration estimates based on actual emissions from all SRS sources for calendar year 1990 plus maximum potential emissions for sources permitted through December 1992.

Table 5-5. Incremental air quality pollutant emission rates related to spent nuclear fuel alternatives - criteria pollutants.[a]

	Baseline		Alternatives						
			No Action	Decentralization			1992/1993 Planning Basis		
Pollutant	Maximum Design Capacity	Actual[b]	1	2a	2b	2c	3a	3b	3c
CRITERIA POLLUTANTS (TONS PER YEAR)									
NO_x	$2.22x10^4$	$2.62x10^3$	—	$6.0x10^0$	$6.0x10^0$	$2.0x10^4$	$6.0x10^0$	$6.0x10^0$	$2.0x10^4$
Particulates									
TSP	$3.62x10^3$	$9.80x10^2$	—	$4.0x10^{-1}$	$4.0x10^{-1}$	$1.5x10^1$	$4.0x10^{-1}$	$4.0x10^{-1}$	$1.5x10^1$
PM_{10}	$2.66x10^3$	$4.97x10^2$	—	$2.6x10^{-1}$	$2.6x10^{-1}$	$9.3x10^0$	$2.6x10^{-1}$	$2.6x10^{-1}$	$9.3x10^0$
CO	$6.77x10^3$	$1.99x10^2$	—	$1.5x10^0$	$1.5x10^0$	$3.8x10^1$	$1.5x10^0$	$1.5x10^0$	$3.8x10^1$
SO_2	$6.42x10^4$	$6.68x10^3$	$1.6x10^{-3}$	$4.0x10^{-1}$	$4.0x10^{-1}$	$1.2x10^1$	$4.0x10^{-1}$	$4.0x10^{-1}$	$1.2x10^1$
Gaseous Fluorides	$2.14x10^{-2}$	$1.07x10^{-2}$	—	—	—	$2.4x10^1$	—	—	$2.4x10^1$
Ozone (as VOC)	N/A[c]	N/A[c]	—	$6.0x10^{-1}$	$6.0x10^{-1}$	$1.8x10^{-1}$	$6.0x10^{-1}$	$6.0x10^{-1}$	$1.8x10^{-1}$

			Regionalization A				Regionalization B		
CRITERIA POLLUTANTS (TONS PER YEAR)			4a	4b	4c	4d	4e	4f	4g
NO_x	$2.22x10^4$	$2.62x10^3$	$8.5x10^0$	$8.5x10^0$	$2.0x10^4$	$1.1x10^1$	$1.1x10^1$	$2.5x10^4$	—
Particulates									
TSP	$3.62x10^3$	$9.80x10^2$	$6.0x10^{-2}$	$6.0x10^{-2}$	$1.5x10^1$	$7.6x10^{-2}$	$7.6x10^{-2}$	$1.5x10^1$	—
PM_{10}	$2.66x10^3$	$4.97x10^2$	$1.45x10^1$	$1.45x10^1$	$9.3x10^0$	$1.8x10^1$	$1.8x10^1$	$9.3x10^0$	—
CO	$6.77x10^3$	$1.99x10^2$	$2.0x10^0$	$2.0x10^0$	$3.8x10^1$	$2.5x10^0$	$2.5x10^0$	$5.2x10^1$	—
SO_2	$6.42x10^4$	$6.68x10^3$	$5.5x10^{-2}$	$5.5x10^{-2}$	$1.3x10^1$	$7.6x10^{-2}$	$7.6x10^{-2}$	$1.7x10^1$	—
Gaseous Fluorides	$2.14x10^{-2}$	$1.07x10^{-2}$	—	—	$2.4x10^1$	—	—	$3.0x10^1$	—
Ozone (as VOC)	N/A[c]	N/A[c]	$8.5x10^{-1}$	$8.5x10^{-1}$	$1.8x10^{-1}$	$1.1x10^0$	$1.1x10^0$	$2.3x10^{-1}$	—

Table 5-5. (continued).

Pollutant	Maximum Design Capacity	Actual[b]	Alternatives			
			Centralization			
CRITERIA POLLUTANTS (TONS PER YEAR)			5a	5b	5c	5d
NO_x	2.2×10^{4}	2.6×10^{3}	5.6×10^{1}	5.6×10^{1}	2.0×10^{4}	—
Particulates						
TSP	3.62×10^{3}	9.8×10^{2}	2.1×10^{0}	2.1×10^{0}	1.8×10^{1}	—
PM_{10}	2.66×10^{3}	4.97×10^{2}	1.4×10^{0}	1.4×10^{0}	9.3×10^{0}	—
CO	6.77×10^{3}	1.99×10^{2}	2.7×10^{1}	2.7×10^{1}	6.9×10^{1}	—
SO_2	6.42×10^{4}	6.68×10^{3}	8.1×10^{0}	8.1×10^{0}	2.0×10^{1}	—
Gaseous Fluorides	2.14×10^{-2}	1.07×10^{-2}			2.4×10^{1}	—
Ozone (as VOC)	N/A[c]	N/A[c]	4.6×10^{0}	4.6×10^{0}	2.4×10^{1}	—

a. Source: WSRC (1994a).

b. Maximum operational air pollutant emissions projected for baseline year 1995. Concentration estimates based on actual emissions from all SRS sources for calendar year 1990 plus maximum potential emissions for sources permitted through December 1992.

c. Emissions data currently unavailable.

— No proposed incremental emissions.

Table 5-6. Incremental air quality pollutant emission rates related to spent nuclear fuel alternatives - toxic pollutants.[a]

Pollutant	Baseline		Alternatives						
	Maximum Design Capacity	Actual[b]	No Action	Decentralization			1992/1993 Planning Basis		
			1	2a	2b	2c	3a	3b	3c
TOXIC POLLUTANTS (TONS PER YEAR)									
Nitric Acid	1.13×10^{3}	2.56×10^{0}	5.1×10^{-2}	5.1×10^{-2}	5.1×10^{-2}	1.24×10^{2}	5.1×10^{-2}	5.1×10^{-2}	1.24×10^{2}
1,1,1-Trichloroethane	8.0×10^{1}	NA[c]	—	—	—	7.02×10^{-1}	—	—	7.02×10^{-1}
Benzene	2.9×10^{1}	4.48×10^{0}	—	—	—	8.02×10^{-1}	—	—	8.02×10^{-1}
Ethanolamine	2.21×10^{-2}	5.35×10^{-3}	1.46×10^{-3}	1.46×10^{-3}	1.46×10^{-3}	1.46×10^{-3}	1.46×10^{-3}	1.46×10^{-3}	1.46×10^{-3}
Ethyl Benzene	2.56×10^{0}	1.07×10^{0}	—	—	—	8.02×10^{-4}	—	—	8.02×10^{-4}
Ethylene Glycol	6.83×10^{-1}	4.17×10^{-1}	2.25×10^{-2}	2.25×10^{-2}	2.25×10^{-2}	4.27×10^{-2}	2.25×10^{-2}	2.25×10^{-2}	4.27×10^{-2}
Formaldehyde	4.55×10^{-2}	4.8×10^{-4}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}
Glycol Ethers	4.36×10^{-3}	1.99×10^{-4}	4.06×10^{-3}	4.06×10^{-3}	4.06×10^{-3}	4.06×10^{-3}	4.06×10^{-3}	4.06×10^{-3}	4.06×10^{-3}
Hexachloronaphthalene	<0.01	NA[c]	3.65×10^{-5}	3.65×10^{-5}	3.65×10^{-5}	3.6×10^{-5}	3.65×10^{-5}	3.65×10^{-5}	3.6×10^{-5}
Hexane	3.54×10^{0}	2.22×10^{-1}	3.28×10^{-3}	3.28×10^{-3}	3.28×10^{-3}	8.13×10^{-1}	3.28×10^{-3}	3.28×10^{-3}	8.13×10^{-1}
Manganese	2.84×10^{-1}	3.43×10^{-1}	—	—	—	1.51×10^{-2}	—	—	1.51×10^{-2}
Methyl Alcohol	6.62×10^{-1}	3.46×10^{-1}	6.84×10^{-2}	6.84×10^{-2}	6.84×10^{-2}	8.68×10^{-2}	6.84×10^{-2}	6.84×10^{-2}	8.68×10^{-2}
Methyl Ethyl Ketone	6.41×10^{0}	3.17×10^{0}	2.19×10^{-3}	2.19×10^{-3}	2.19×10^{-3}	3.47×10^{-2}	2.19×10^{-3}	2.19×10^{-3}	3.47×10^{-2}
Methyl Isobutyl Ketone	8.25×10^{0}	2.25×10^{0}	—	—	—	1.27×10^{-2}	—	—	1.27×10^{-2}
Methylene Chloride	1.53×10^{0}	1.19×10^{0}	—	—	—	8.23×10^{-1}	—	—	8.23×10^{-1}
Naphthalene	7.22×10^{-2}	3.08×10^{-2}	5.84×10^{-4}	5.84×10^{-4}	5.84×10^{-4}	6.08×10^{-4}	5.84×10^{-4}	5.84×10^{-4}	6.08×10^{-4}
Phenol	8.07×10^{-2}	1.37×10^{-2}	—	—	—	6.01×10^{-5}	—	—	6.01×10^{-5}
Phosphorus	2.97×10^{-3}	1.65×10^{-4}	—	—	—	1.6×10^{-6}	—	—	1.6×10^{-6}
Sodium Hydroxide	1.26×10^{-1}	1.26×10^{-1}	—	—	—	5.97×10^{-2}	—	—	5.97×10^{-2}
Toluene	3.91×10^{0}	7.66×10^{-1}	5.0×10^{-2}	5.0×10^{-2}	5.0×10^{-2}	9.2×10^{-1}	5.0×10^{-2}	5.0×10^{-2}	9.2×10^{-1}
Trichloroethylene	2.52×10^{1}	9.8×10^{0}	—	—	—	5.52×10^{-4}	—	—	5.52×10^{-4}
Vinyl Acetate	4.38×10^{-2}	5.9×10^{-3}	—	—	—	5.0×10^{-5}	—	—	5.0×10^{-5}
Xylene	1.46×10^{3}	1.22×10^{1}	1.58×10^{-1}	1.58×10^{-1}	1.58×10^{-1}	1.4×10^{0}	1.58×10^{-1}	1.58×10^{-1}	1.4×10^{0}

Table 5-6. (continued).

Pollutant	Baseline: Maximum Design Capacity	Baseline: Actual[b]	Alternatives: Regionalization A: 4a	4b	4c	Regionalization B: 4d	4e	4f	4g
TOXIC POLLUTANTS (TONS PER YEAR)									
Nitric Acid	1.1×10^{3}	2.6×10^{0}	5.1×10^{-2}	5.1×10^{-2}	1.2×10^{2}	6.5×10^{-2}	6.5×10^{-2}	1.5×10^{2}	—
1,1,1-Trichloroethane	8.0×10^{1}	NA[c]	—	—	7.0×10^{-1}	—	—	8.9×10^{-1}	—
Benzene	2.9×10^{1}	4.5×10^{0}	—	—	8.0×10^{-1}	—	—	1.0×10^{0}	—
Ethanolamine	2.2×10^{-2}	5.4×10^{-3}	1.5×10^{-3}	1.5×10^{-3}	1.5×10^{-3}	1.9×10^{-3}	1.9×10^{-3}	1.9×10^{-3}	—
Ethyl Benzene	2.6×10^{0}	1.1×10^{0}	—	—	8.0×10^{-4}	—	—	1.0×10^{-3}	—
Ethylene Glycol	6.8×10^{-1}	4.2×10^{-1}	2.3×10^{-2}	2.3×10^{-2}	4.3×10^{-2}	2.9×10^{-2}	2.9×10^{-2}	5.5×10^{-2}	—
Formaldehyde	4.6×10^{-2}	4.8×10^{-4}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-5}	4.6×10^{-6}	4.6×10^{-6}	4.6×10^{-6}	—
Glycol Ethers	4.4×10^{-3}	2.0×10^{-4}	4.1×10^{-3}	4.1×10^{-3}	4.1×10^{-3}	5.2×10^{-3}	5.2×10^{-3}	5.2×10^{-3}	—
Hexachloronapthalene	<0.01	NA[c]	3.7×10^{-5}	3.7×10^{-5}	3.6×10^{-5}	4.7×10^{-5}	4.7×10^{-5}	4.6×10^{-5}	—
Hexane	3.5×10^{0}	2.2×10^{-1}	3.3×10^{-3}	3.3×10^{-3}	8.1×10^{-1}	4.2×10^{-3}	4.2×10^{-3}	1.0×10^{0}	—
Manganese	2.8×10^{-1}	3.4×10^{-1}	—	—	1.5×10^{-2}	—	—	1.9×10^{-2}	—
Methyl Alcohol	6.6×10^{-1}	3.5×10^{-1}	6.8×10^{-2}	6.8×10^{-2}	8.7×10^{-2}	8.6×10^{-2}	8.6×10^{-2}	1.1×10^{-1}	—
Methyl Ethyl Ketone	6.4×10^{0}	3.2×10^{0}	2.2×10^{-3}	2.2×10^{-3}	3.5×10^{-2}	2.8×10^{-3}	2.8×10^{-3}	4.4×10^{-2}	—
Methyl Isobutyl Ketone	8.3×10^{0}	2.3×10^{0}	—	—	1.3×10^{-2}	—	—	1.7×10^{-2}	—
Methylene Chloride	1.5×10^{0}	1.2×10^{0}	—	—	8.2×10^{-1}	—	—	1.0×10^{0}	—
Naphthalene	7.2×10^{-2}	3.1×10^{-2}	5.8×10^{-4}	5.8×10^{-4}	6.1×10^{-4}	7.4×10^{-4}	7.4×10^{-4}	7.7×10^{-4}	—
Phenol	8.1×10^{-2}	1.4×10^{-2}	—	—	6.0×10^{-5}	—	—	7.6×10^{-5}	—
Phosphorus	3.0×10^{-3}	1.7×10^{-4}	—	—	1.6×10^{-6}	—	—	2.0×10^{-6}	—
Sodium Hydroxide	1.3×10^{-1}	1.3×10^{-1}	—	—	6.0×10^{-2}	—	—	7.6×10^{-2}	—
Toluene	3.9×10^{0}	7.7×10^{-1}	5.0×10^{-2}	5.0×10^{-2}	9.2×10^{-1}	6.4×10^{-2}	6.4×10^{-2}	1.2×10^{0}	—
Trichloroethylene	2.5×10^{1}	9.8×10^{0}	—	—	5.5×10^{-4}	—	—	7.0×10^{-4}	—
Vinyl Acetate	4.4×10^{-2}	5.9×10^{-3}	—	—	5.0×10^{-5}	—	—	6.4×10^{-5}	—
Xylene	1.5×10^{3}	1.2×10^{1}	1.6×10^{-1}	1.6×10^{-1}	1.4×10^{0}	2.0×10^{-1}	2.0×10^{-1}	1.8×10^{0}	—

Table 5-6. (continued).

Pollutant	Maximum Design Capacity	Actual[b]	Alternatives: Centralization 5a	5b	5c	5d
TOXIC POLLUTANTS (TONS PER YEAR)						
Nitric Acid	1.1×10^{3}	2.6×10^{0}	5.1×10^{-2}	5.1×10^{-2}	1.2×10^{2}	—
1,1,1-Trichloroethane	8.0×10^{1}	NA[c]	—	—	7.0×10^{-1}	—
Benzene	2.9×10^{1}	4.5×10^{0}	—	—	8.0×10^{-1}	—
Ethanolamine	2.2×10^{-2}	5.4×10^{-3}	1.5×10^{-3}	1.5×10^{-3}	1.5×10^{-3}	—
Ethyl Benzene	2.6×10^{0}	1.1×10^{0}	—	—	8.0×10^{-4}	—
Ethylene Glycol	6.8×10^{-1}	4.2×10^{-1}	2.3×10^{-2}	2.3×10^{-2}	4.3×10^{-2}	—
Formaldehyde	4.6×10^{-2}	4.8×10^{-4}	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}	—
Glycol Ethers	4.4×10^{-3}	2.0×10^{-4}	4.1×10^{-3}	4.1×10^{-3}	4.1×10^{-3}	—
Hexachloronapthalene	<0.01	NA[c]	3.7×10^{-5}	3.7×10^{-5}	3.6×10^{-5}	—
Hexane	3.5×10^{0}	2.2×10^{-1}	3.3×10^{-3}	3.3×10^{-3}	8.1×10^{-1}	—
Manganese	2.8×10^{-1}	3.4×10^{-1}	—	—	1.5×10^{-2}	—
Methyl Alcohol	6.6×10^{-1}	3.5×10^{-1}	6.8×10^{-2}	6.8×10^{-2}	8.7×10^{-2}	—
Methyl Ethyl Ketone	6.4×10^{0}	3.2×10^{0}	2.2×10^{-3}	2.2×10^{-3}	3.5×10^{-2}	—
Methyl Isobutyl Ketone	8.3×10^{0}	2.3×10^{0}	—	—	1.3×10^{-2}	—
Methylene Chloride	1.5×10^{0}	1.2×10^{0}	—	—	8.2×10^{-1}	—
Naphthalene	7.2×10^{-2}	3.1×10^{-2}	5.8×10^{-4}	5.8×10^{-4}	6.1×10^{-4}	—
Phenol	8.1×10^{-2}	1.4×10^{-2}	—	—	6.0×10^{-5}	—
Phosphorus	3.0×10^{-3}	1.7×10^{-4}	—	—	1.6×10^{-6}	—
Sodium Hydroxide	1.3×10^{-1}	1.3×10^{-1}	—	—	6.0×10^{-2}	—
Toluene	3.9×10^{0}	7.7×10^{-1}	5.0×10^{-2}	5.0×10^{-2}	9.2×10^{-1}	—
Trichloroethylene	2.5×10^{1}	9.8×10^{0}	—	—	5.5×10^{-4}	—
Vinyl Acetate	4.4×10^{-2}	5.9×10^{-3}	—	—	5.0×10^{-5}	—
Xylene	1.5×10^{3}	1.2×10^{1}	1.6×10^{-1}	1.6×10^{-1}	1.4×10^{0}	—

a. Source: WSRC (1994a).
b. Maximum operational air pollutant emissions projected for baseline year 1995. Concentration estimates based on actual emissions from all SRS sources for calendar year 1990 plus maximum potential emissions for sources permitted through December 1992.
c. NA= Emissions data currently unavailable.
— No proposed incremental emissions.

Radiological Emissions. DOE evaluated the potential radiological releases to the atmosphere from spent fuel management at the SRS using existing Site historical operations information. Based on the actual 1993 emissions data from the Receiving Basin for Offsite Fuels (WSRC 1994d), DOE estimates that emissions from any of the wet storage options under Alternatives 1 through 4 would consist of about 2×10^{-7} curies per year of cesium-137. Releases from dry storage activities under these alternatives would be somewhat less. For Alternative 5 where SRS would manage about 2,760 MTHM (3,042 tons) of spent fuel (versus about 200 to 250 MTHM [220 to 276 tons] for the other alternatives), the atmospheric releases of cesium-137 would be proportionally higher.

DOE used actual emissions from F- and H-Areas during 1985 and 1986, a period when the SRS was processing material through the separations facilities at close to maximum capacity to evaluate potential releases during processing of the spent nuclear fuel. DOE believes that the isotopes released during this period, and their emission rates, are representative of emissions that could occur during the processing under any of the alternatives, (Table 5-7). The results of the analyses are presented in this section and the human health consequences are discussed in Section 5.12. Section 5.15 presents the analysis of the consequences of accidents.

Construction Emissions. Potential impacts to air quality from construction activities would include fugitive dust from the clearing of land, as well as exhaust emissions from support equipment (e.g., earth-moving vehicles, diesel generators). The amount of dust produced would be proportional to the land area disturbed for the new facilities, all of which would be located near the center of the Site. The areas affected by each alternative would be as follows:

- No Action - 0 acres
- Decentralization, 1992/1993 Planning Basis and Regionalization A (by fuel type) - 6 to 9 acres
- Regionalization B (by location) - 7 to 11 acres
- Centralization - 40 to 100 acres
- Shipping fuel offsite - 1 acre

Table 5-7. Estimated incremental annual emissions in curies of radionuclides to the atmosphere from processing under each alternative.

Radionuclide	SRS Baseline[a, b]
Tritium (elemental)	1.88×10^{5}
Cesium-134	3.60×10^{-4}
Cesium-137	4.07×10^{-3}
Curium-244	2.00×10^{-4}
Cerium-141	1.83×10^{-3}
Cerium-144	3.11×10^{-2}
Amercurium-241	2.27×10^{-4}
Cobalt-60	4.00×10^{-6}
Plutonium-238	1.28×10^{-3}
Plutonium-239	4.01×10^{-4}
Strontium-90	1.39×10^{-2}
Rubidium-103	7.25×10^{-3}
Uranium-235	2.00×10^{-3}
Osmium-185	3.60×10^{-4}
Nibium-95	2.89×10^{-2}
Selenium-75	1.52×10^{-5}
Zirconium-95	1.68×10^{-2}
Rubidium-106	5.12×10^{-3}
Krypton-85	6.80×10^{5}
Carbon-14	2.80×10^{1}

a. Source: Hamby to Shedrow, 12/13/93.
b. Source terms are taken from 1985/86 F/H Area releases.

DOE anticipates that overall construction impacts to air quality would be minimal and of a short duration (6 months to 3 years). The SRS sitewide compliance with state and Federal ambient air quality standards would not be affected by any construction-related activities associated with spent fuel management.

5.7.1 Alternative 1 - No Action

The SRS would not process any spent nuclear fuel under the No Action alternative. Normal site baseline emissions would continue (Tables 5-3, 5-4, 5-5, 5-6 and 5-7). DOE would not construct any new facilities under this alternative.

5.7.2 Alternative 2 - Decentralization

Atmospheric emissions under two of the Decentralizations options (dry storage and wet storage) would be similar to those for No Action. Those from the processing of the spent fuel (Option 2c) would be of somewhat higher concentrations (Tables 5-3, 5-4, 5-5, 5-6 and 5-7). The emissions would originate from existing facilities involved in the management of spent fuel under this alternative as well as new ones that DOE would construct (Figure 3-2).

5.7.3 Alternative 3 - 1992/1993 Planning Basis

Emissions to the atmosphere would be similar to those for Alternative 2 because the amount of fuel managed would be similar [216 and 211 MTHM (238 and 233 tons), Alternative 3 and Alternative 2 respectively] and the facilities required would be the same (Figure 3-2).

5.7.4 Alternative 4 - Regionalization

Regionalization A (by fuel type). Atmospheric emissions would be similar to the releases from Alternative 2 because of the similarity in volumes of fuel managed [208 and 211 MTHM (239 and 233 tons), respectively] and in the facilities involved (Figure 3-2).

Regionalization B (by location). Emissions would be somewhat higher than for Regionalization A for both dry and wet storage options if the SRS receives all the spent fuel in the eastern portion of the country, because the Site would manage about 19 percent more fuel. Atmospheric emissions from processing would not change from those under other alternatives because the amount of aluminum-clad fuel involved would be the same. Facility requirements would also be similar (Figure 3-2).

Shipping all of the current SRS inventory off the Site (Option 4g) would result in the lowest emissions to the atmosphere of any of the options under this alternative. These releases would result from the characterization and canning of the fuel prior to shipment.

5.7.5 Alternative 5 - Centralization

The atmospheric emissions resulting from centralizing all the spent nuclear fuel at the SRS would be the greatest of all the alternatives. The Site would manage about 2,760 MTHM (3,042 tons) of fuel. Releases from storage activities for centralization would be proportionally higher than for the other alternatives where the SRS would manage about 200 to 250 MTHM (220 to 276 tons) of spent fuel. However, emissions from processing under Alternative 5 would be similar to those under the other alternatives because the same amount of aluminum-clad fuel would be processed in each case. The facilities required under all three options would be similar in function (Figure 3-2) but of much larger capacity than for other alternatives.

Shipping all the SRS fuel to another site (Option 5d) would result in the lowest level of atmospheric releases of any alternative, similar to those under Regionalization B, Option 4g.

5.8 Water Quality and Related Consequences

SRS use of surface-water and groundwater resources under any of the alternatives would not substantially increase the volumes currently used for process, cooling, and domestic water on the Site. Table 5-8 summarizes the groundwater and surface water usage requirements for each alternative and option, and compares them to current SRS usages.

The Centralization Alternative (Option 5c), under which DOE would transfer all spent nuclear fuel to the SRS, would result in the largest amount of water use [approximately 378.5 million liters (100 million gallons) per year], which is a small amount compared to current SRS water requirements of approximately 88.2 billion liters (23.3 billion gallons) per year. This represents an increase of approximately 0.4 percent above current usage. Therefore, DOE anticipates that water use under any of the alternatives would have minimal impact on the water resources of the Site.

The impact on water quality of the operation of any of the alternatives would also be minimal. Existing SRS treatment facilities could accommodate all new spent fuel-related domestic and process

Table 5-8. Annual groundwater and surface water usage requirements for each alternative.[a,b]

Alternative	Groundwater Usage per Year	Surface Water Usage per Year	Total Annual
Current SRS Usage	12.5 billion liters	75.7 billion liters	88.2 billion liters
No Action			
Option 1 - Wet Storage	35.1 million liters	None	35.1 million liters
Decentralization			
Option 2a - Dry Storage	48.7 million liters	6.1 million liters	54.8 million liters
Option 2b - Wet Storage	50.6 million liters	7.2 million liters	57.8 million liters
Option 2c - Processing[c]	48.7 million liters	310.8 million liters	359.5 million liters
Planning Basis			
Option 3a - Dry Storage	48.7 million liters	6.1 million liters	54.8 million liters
Option 3b - Wet Storage	50.6 million liters	7.2 million liters	57.8 million liters
Option 3c - Processing[c]	48.7 million liters	310.8 million liters	359.5 million liters
Regionalization - A			
Option 4a - Dry Storage	48.7 million liters	6.1 million liters	54.8 million liters
Option 4b - Wet Storage	50.6 million liters	7.2 million liters	57.8 million liters
Option 4c - Processing[c]	47.6 million liters	308.8 million liters	356.5 million liters
Regionalization - B			
Option 4d - Dry Storage	48.7 million liters	6.1 million liters	54.8 million liters
Option 4e - Wet Storage	50.6 million liters	7.2 million liters	57.8 million liters
Option 4f - Processing[c]	48.7 million liters	310.8 million liters	356.5 million liters
Option 4g - Ship Out[c]	38.1 million liters	3.0 million liters	41.1 million liters
Centralization			
Case 5a - Dry Storage	67.7 million liters	6.1 million liters	73.8 million liters
Case 5b - Wet Storage	69.6 million liters	7.2 million liters	76.8 million liters
Case 5c - Processing[c]	67.7 million liters	310.8 million liters	378.5 million liters
Case 5d - Ship Out[c]	38.1 million liters	3.0 million liters	41.1 million liters

a. Source: WSRC (1994b).
b. To convert liters to gallons, multiply by 0.26418.
c. First 10 years only.

wastewater streams. The expected total SRS flow volumes would still be well within the design capacities of the Site treatment systems. Because these plants would continue to meet National Pollutant Discharge Elimination System limits and reporting requirements, DOE expects no impact on the water quality of the receiving streams. The increased cooling water flows would also meet all discharge permit limits and would have minimal impacts on the receiving water.

Each of the alternatives would contribute to the very small releases of radionuclides that normal SRS operations discharge to the surface water through federally permitted wastewater outfalls. Table 5-9 summarizes the estimated maximum amounts of radioactivity that could be released to the Savannah River in the form of the five isotopes that DOE anticipates could be present in liquid

Table 5-9. Estimated maximum liquid radiological releases to the Savannah River for all alternatives.

Alternative	Released Amount (Ci/yr)				
	Tritium-3	Strontium-90	Iodine-129	Cesium-137	Plutonium-239
No Action					
Option 1 - Wet Storage	1.3×10^{4}	2.4×10^{-1}	2.2×10^{-2}	1.1×10^{-1}	7.0×10^{-3}
Decentralization					
Option 2a - Dry Storage	1.4×10^{4}	2.5×10^{-1}	2.2×10^{-2}	1.1×10^{-1}	7.1×10^{-3}
Option 2b - Wet Storage	1.4×10^{4}	2.5×10^{-1}	2.2×10^{-2}	1.1×10^{-1}	7.1×10^{-3}
Option 2c - Processing	1.4×10^{4}	2.5×10^{-1}	2.2×10^{-2}	1.1×10^{-1}	7.1×10^{-3}
Planning Basis					
Option 3a - Dry Storage	1.4×10^{4}	2.6×10^{-1}	2.3×10^{-2}	1.1×10^{-1}	7.4×10^{-3}
Option 3b - Wet Storage	1.4×10^{4}	2.6×10^{-1}	2.3×10^{-2}	1.1×10^{-1}	7.4×10^{-3}
Option 3c - Processing	1.4×10^{4}	2.6×10^{-1}	2.3×10^{-2}	1.1×10^{-1}	7.4×10^{-3}
Regionalization - A					
Option 4a - Dry Storage	1.3×10^{4}	2.4×10^{-1}	2.2×10^{-2}	1.0×10^{-1}	6.9×10^{-3}
Option 4b - Wet Storage	1.3×10^{4}	2.4×10^{-1}	2.2×10^{-2}	1.0×10^{-1}	6.9×10^{-3}
Option 4c - Processing	1.3×10^{4}	2.4×10^{-1}	2.2×10^{-2}	1.0×10^{-1}	6.9×10^{-3}
Regionalization - B					
Option 4d - Dry Storage	1.7×10^{4}	3.1×10^{-1}	2.8×10^{-2}	1.3×10^{-1}	9.0×10^{-3}
Option 4e - Wet Storage	1.7×10^{4}	3.1×10^{-1}	2.8×10^{-2}	1.3×10^{-1}	9.0×10^{-3}
Option 4f - Processing	1.7×10^{4}	3.1×10^{-1}	2.8×10^{-2}	1.3×10^{-1}	9.0×10^{-3}
Option 4g - Ship Out	$<1.3 \times 10^{4}$	$<2.4 \times 10^{-1}$	$<2.2 \times 10^{-2}$	$<1.1 \times 10^{-1}$	$<7.0 \times 10^{-3}$
Centralization					
Case 5a - Dry Storage	1.8×10^{5}	3.2	2.9×10^{-1}	1.4	9.2×10^{-2}
Case 5b - Wet Storage	1.8×10^{5}	3.2	2.9×10^{-1}	1.4	9.2×10^{-2}
Case 5c - Processing	1.8×10^{5}	3.2	2.9×10^{-1}	1.4	9.2×10^{-2}
Case 5d - Ship Out	$<1.3 \times 10^{4}$	$<2.4 \times 10^{-1}$	$<2.2 \times 10^{-2}$	$<1.1 \times 10^{-1}$	$<7.0 \times 10^{-3}$

effluents from normal spent nuclear fuel management activities. These estimates are based on releases to surface waters reported for the Site for the 1985 to 1986 period when the F- and H-Areas were operating at or near capacity. Consequently, the estimated maximum liquid radiological releases given in Table 5-9 represent the upper limits of what DOE anticipates could be released by each alternative, and are therefore very conservative values for wet and dry storage. In all cases, the concentrations of radionuclides will continue to be well within dose limits established by DOE.

The consequences to human health due to these releases are discussed in Section 5.12, Occupational and Public Health and Safety.

Construction of new facilities under any alternative would require amounts of water that would be only a very small percentage of the current daily water use at the SRS. Good engineering practice measures would prevent sediment runoff or spills of fuel or chemicals. Therefore, construction activities should have no impact on the quality surface or groundwater at the Site.

DOE also analyzed the potential impacts of accidents in F- and H-Areas on ground- and surface-water quality. The analysis evaluated two types of accidental releases: one to the ground surface (e.g., overflow of a wet storage pool) and another directly to the subsurface (e.g., failure of a pool liner). Because pool water could contain some radionuclides, but would not contain any toxic or harmful chemicals, the following evaluation addresses only the consequences of radionuclide releases.

A release of pool water onto the ground from the Receiving Basin for Offsite Fuels, in H-Area, would not flow directly into any stream or other surface-water body. The building is in a graded, gravel-covered area among other buildings and alongside a railroad spur and access road. A tank farm surrounded by an earthen berm is immediately to the south. A channelized drainage ditch begins approximately 244 meters (800 feet) west of the basin building and passes through culverts under a railroad line and Road E before emptying into a tributary of Fourmile Branch about 500 meters (1,650 feet) from the Receiving Basin. The grading at the Site would contain a small volume of water overflowing the basin in the immediate area of the building. In the unlikely event that a larger spill reached the drainage ditch to the west, DOE could contain the water by blocking either of the two culverts through which the drainage ditch passes. After containing the spilled water, DOE could remove and properly dispose of it. DOE would design and construct new facilities containing storage pools in a manner that would confine any overflow or other surface release of pool water. Therefore, DOE believes that there will be no direct release to surface water from spills of pool water at an existing or potential facility.

An overflow from a pool could reach the groundwater by slowly flowing downward from the surface through the unsaturated zone until it reached the water table, which is 9 to 15 meters (30 to 50 feet) below the grade in the F- and H-Areas. Overflow water would take several years to reach the water table, based on a vertical velocity of between 0.9 and 2.1 meters (3 to 7 feet) per year (DOE 1987). As discussed in the following paragraphs, once in the groundwater, a plume would take many years to reach either of the closest surface-water bodies, Fourmile Branch to the south or Upper Three Runs Creek to the north.

DOE has calculated the travel times of groundwater in the F- and H-Areas based on specific information on the hydraulic conductivity, the hydraulic gradient, and the effective porosity of aquifers in this area (WSRC 1993c) and on the use of Darcy's Law. Water would take between 16 and 500 years to travel 1.6 kilometers (1 mile) toward Fourmile Branch or Upper Three Runs Creek. These estimates of travel time agree with values obtained from the results of DOE modeling studies performed on the F- and H-Areas (Geotrans 1993; appended to WSRC 1993c). The reason for this wide range of potential travel time is that the hydraulic conductivity of aquifer materials is highly variable and can vary in the same aquifer by several orders of magnitude. This slow movement through the subsurface, either vertically through the unsaturated zone or horizontally within the aquifer, would facilitate the removal of radionuclides from the spill plume through a number of processes. These include radioactive decay, trapping of particulates in the soil, and ion exchange and adsorption by the soil (Hem 1989). DOE believes that travel time of a contaminant plume through the subsurface in the F- or H-Area or in the adjacent representative host site would be such that no radionuclides would reach Fourmile Branch, Upper Three Runs Creek, or any other surface-water body by this route. For the same reasons, no radioactive contaminants introduced into the subsurface in these areas would move off the Site in groundwater.

DOE does not believe that releases of radionuclides such as those described above would reach SRS drinking-water sources that lie in deep aquifers under the Site. These aquifers are several hundred feet below the ground surface, and a number of thick aquifers and aquitards separate them from the water table aquifer (see Section 4.8). In addition to the distances and the presence of confining layers, vertical flow in the intervening stratified sedimentary aquifers is slow in comparison to horizontal flow. Radionuclide contamination of offsite drinking water sources is even more unlikely given the depth of their source aquifers, the distances involved, and the attenuation of contaminants in the soils, as described above.

DOE also evaluated a second kind of unintentional release in the F- or H-Area, a direct leak to the subsurface from a breach in a storage pool during routine operations. The analysis assumed a 19-liter (5-gallon)-per-day leak as a result of secondary containment or piping failure at a new state-of-the-art wet storage and fuel transfer facility. The analysis assumed further that the leak would go undetected for 1 month, a conservative assumption given the sensitivity of the leak detection equipment that these new facilities would require. The reliability and sensitivity of the leak detection devices would be equal to or superior to those required by the U.S. Nuclear Regulatory Commission (NRC 1975) for spent nuclear fuel storage facilities in commercial nuclear power plants. DOE would require spent nuclear fuel storage pools (whether fuel unloading pools or storage basins) to have leak detection monitoring devices, pool water level monitors, and radiation monitors designed to alarm both locally and in a continuously staffed central location.

To provide a common basis for analysis of spent nuclear fuel alternatives at its various sites, DOE developed a generic infrastructure design for a hypothetical spent nuclear fuel complex (Hale 1994). This design includes proposed criteria for temporary wet storage basins, fuel loading and unloading pools, and transfer canals.

Based on these design criteria, a leak from one of these basins if constructed in the F- or H-Area could result in the introduction of radionuclide-contaminated water into the ground at depths as much as 13 meters (43 feet) below grade. Such a release would go directly to the water table aquifer or to the unsaturated zone above it, depending on the depth of the water table. In either case, the processes governing the slow plume movement (i.e., the hydraulic conductivity, hydraulic gradient, and effective porosity of aquifers in the F- and H-Areas) and the processes resulting in the attenuation of contaminants and radionuclides (i.e., radioactive decay, trapping of particulates in the soil, ion exchange in the soil, and adsorption to soil particles) described in the previous paragraphs would also prevent (or at least mitigate) impacts to surface-or groundwater resources from releases of this type. There could be localized contamination of groundwater in the surface aquifer in the immediate vicinity of the storage facilities. This aquifer is not used as a source of drinking water. DOE believes that no radionuclide contamination of deeper confined aquifers that are sources of onsite or offsite drinking water could occur from a release of this type.

DOE is currently evaluating potential water pathways that might result from releases from K- and L-Reactor basins.

5.8.1 Alternative 1 - No Action

5.8.1.1 Option 1 - Wet Storage. During operations under this alternative, current levels of water usage would not change. Nor would changes occur in thermal discharges from cooling water or the quantity or quality of radioactive and nonradioactive wastewater effluents.

The viable accidents under this alternative would be a release of pool water onto the ground surface or a breach of the liner of the wet storage basins in which the spent nuclear fuel would be stored. As discussed above, radionuclides in the released water would enter the water table aquifer but would not reach any surface-water or any drinking water aquifer on or off the SRS. Basin water contains no toxic or hazardous chemicals. Therefore, accidental releases from the basins would have minimal impacts on surface- and groundwater resources.

Spills of chemicals would not reach surface- or groundwater due to existing proper engineering design and environmental controls, and to rapid containment and cleanup.

5.8.2 Alternative 2 - Decentralization

Operations under either the dry or wet storage option for the Decentralization alternative would increase Site water usage by less than 0.1 percent above current levels. Processing would increase use by about 0.4 percent. Release of nonradioactive and radioactive materials to surface waters would increase only slightly and would be well within discharge permit limits and DOE dose limits. There would be no releases to groundwater during normal operations. Overall impacts to water quantity and water quality would be minimal.

Impacts to water resources due to accidental releases onto the ground or into the subsurface would also be minimal as explained above. Potential contamination would be limited to the surface aquifer.

5.8.3 Alternative 3 - 1992/1993 Planning Basis

DOE expects that the impacts to water resources under the three dry storage, wet storage, and processing cases for this alternative would be similar to those described for the same options under Alternative 2, Decentralization. Overall impacts would be minimal.

5.8.4 Alternative 4 - Regionalization

DOE expects that the impacts to water resources under the three options for regionalization by fuel type (Regionalization A) would be similar to those described for the same options under Alternative 2, Decentralization. Regionalization B (by geographic location) would result in impacts somewhat greater than those for Alternative 2 because the SRS would have to manage an additional 55 MTHM (61 tons) of spent fuel. In either case, overall impacts would be minimal. For Option 4g, shipping all SRS fuel to Oak Ridge Reservation, impacts to water resources would be the smallest of any alternative, similar to those for Option 5d - Centralization.

5.8.5 Alternative 5 - Centralization

The first three options for this alternative - dry storage (Option 5a), wet storage (Option 5b), and processing (Option 5c) - assume that DOE would transfer all spent nuclear fuel to the SRS for management. The impacts of operations to water resources under these options would be similar in nature to the impacts for the same options under Alternative 2, Decentralization, as described in Section 5.8.2. However, the extent of the impacts would be greater because the number and size of facilities that DOE would construct and operate and the quantities of fuel it would manage would be larger than those for any other alternative. Even so, DOE expects that the overall impacts of construction and operation to be minor. For example, the total volume of water that the SRS would withdraw for construction, cooling, processing, and domestic use under any of these three options would not exceed approximately 378.5 million liters (100 million gallons) per year. This requirement would be approximately 0.4 percent of the 88.2 billion liters (23.3 billion gallons) that the SRS currently uses annually.

Similarly, DOE believes that the overall impacts of accidents under any of these three options would be minor, even though the number and size of the facilities would be greater under this alternative than for any other. Radionuclides released during an accident would not affect any surface-water or any drinking water aquifer. However, surface aquifer resources would receive contamination in the area of any release.

For Option 5d (shipping the spent nuclear fuel off the Site), impacts to water resources would be smaller than those for any other alternative or option. DOE would have to build only one new facility (for fuel characterization) and the spent fuel would remain at SRS only for the first part of the 40-year management period. Overall impacts would be minimal.

5.9 Ecology

DOE expects that construction impacts, which would include loss of some wildlife habitat due to land clearing, would be greatest under the Centralization Alternative, Dry Storage option. Representative impacts from operations would include disturbance and displacement of animals caused by movement and noise of personnel, equipment, and vehicles; however, these impacts would be minor under all the proposed alternatives. Construction and operation would not disturb any critical or sensitive habitat, nor would they affect any wetland areas. DOE anticipates that the proposed alternatives would have minimal impacts on the SRS flora and fauna from the transport of radionuclides.

5.9.1 Alternative 1 - No Action

Under this alternative, DOE could refurbish or modify existing wet storage facilities and would confine any activity to these facilities. As a consequence, DOE expects no impacts to ecological resources. Impacts of operations under this alternative would be minimal, limited to some minor disturbance of animals by vehicular traffic.

5.9.2 Alternative 2 - Decentralization

5.9.2.1 Option 2a - Dry Storage. This option would require some new construction, but any construction activity would occur either within the boundaries of F- or H-Area, which is already heavily developed, or adjacent to it. As a result, this construction would have little or no impact on ecological resources. There would be no impacts to wetlands, threatened or endangered species, socially or commercially important species (such as the eastern wild turkey), or disturbance-sensitive species (such as wood warblers and vireos). Impacts of operations under this option would be limited to some minor disturbance of animals by slight increases in vehicular traffic. No threatened, endangered, or candidate species occur in the area of operations. Species likely to be disturbed or killed by vehicles (e.g., cotton rat, gray squirrel, opossum, and white-tailed deer) are common to ubiquitous in the area. Overall impact to ecological resources would be minimal.

5.9.2.2 Option 2b - Wet Storage. Construction impacts would be similar to those described for dry storage (Option 2a). Impacts of operations under this option would also be similar to those described for dry storage (Option 2a). Overall impacts to ecological resources would be minimal.

5.9.2.3 Option 2c - Processing and Storage. Construction and operations impacts for this option would also be similar to those for dry storage (Option 2a). Overall impacts would still be minimal.

5.9.3 Alternative 3 - 1992/1993 Planning Basis

Both construction and operational impacts for the three options under this alternative would be similar to those described for Alternative 2 - Decentralization. Overall impacts would be minimal.

5.9.4 Alternative 4 - Regionalization

Under the Regionalization A alternative, impacts to ecological resources would be minimal as described for Alternative 2. Impacts due to the Regionalization B options would be somewhat greater due to the larger volume of spent fuel that the SRS would manage. Overall impacts would still be minimal, however.

The smallest impacts would occur under Option 4g because DOE would ship all spent fuel off the Site.

5.9.5 Alternative 5 - Centralization

5.9.5.1 Option 5a - Dry Storage. The discussion that follows assumes that any facility development would take place in an area that does not contain any pristine wetlands, old growth timber, threatened and endangered species, or designated critical habitat. More specifically, because the upland areas south and east of H-Area are dominated by planted pine (primarily loblolly and slash) stands, the discussion of impacts assumes that any facility development in support of spent nuclear fuel management would take place in an area of 5- to 40-year-old pines. Finally, the analysis assumes that any facility development would require a site-specific National Environmental Policy Act review as required under 10 CFR Part 1021 and in accordance with the Council on Environmental Quality's NEPA implementing regulations (40 CFR Parts 1500-1508).

The proposed interim dry storage facility and support facilities, requiring approximately 0.28 square kilometers (70 acres) to 0.4 square kilometer (100 acres) of land, would be built somewhere within the largely wooded roughly 2.8 square kilometer (700-acre) area south and east of H-Area west of F-Road, and north of Fourmile Branch. This area has a number of advantages; among them: it would be relatively easy to connect with existing utilities (gas, water, sewer); it would minimize the amount of supporting infrastructure (e.g., railroad spurs, access roads, and transmission lines) that would have to be built; and it would enable DOE to consolidate spent nuclear fuel management activities in an area that has been altered many times over the years by farming (before 1951) and timber management activities (after 1951).

Construction activities would result in the clearing of as much as approximately 0.4 square kilometer (100 acres) of planted 5- to 40-year-old loblolly or slash pine for new facilities on the undeveloped representative host site south and east of H-Area. This land clearing would involve a relatively small number of loggers and heavy equipment operators, but probably would drive most birds and larger, more mobile animals from the area. Some smaller, less mobile animals, such as turtles, toads, lizards, mice, and voles, probably would be killed. Aside from the loss of 0.28 to 0.4 square kilometer (70-100 acres) of planted pines that provide habitat for a limited number of reptiles, birds, and mammals, construction impacts would be minor.

Any land clearing and timber harvesting conducted on the undeveloped host site would be carefully planned and conducted according to widely accepted Best Management Practices to minimize erosion and soil loss and to prevent impacts to downgradient wetlands and streams. DOE and SRS policy is to achieve "no net loss" of wetlands. DOE has issued a guidance document, *Information for Mitigation of Wetland Impacts at the Savannah River Site* (DOE 1992), for project planners that puts forth a practical approach to wetlands protection that begins with avoidance of impacts (if possible), moves to minimization of impacts (if avoidance is impossible), and requires compensatory mitigation (wetlands restoration, creation, enhancement, or acquisition) in the event that impacts cannot be avoided.

In the event that new facility development was required, DOE would perform predevelopment surveys to ensure that its activities would not affect threatened and endangered species or sensitive habitats. To the extent practicable, land clearing and timber harvesting would be restricted to times of the year when songbirds and game birds were not nesting or rearing young. In South Carolina, most songbirds nest, rear, and fledge young from March to September (Sprunt and Chamberlain 1970). Quail, dove, and wild turkey in the region normally nest and fledge young during the spring and summer (Sprunt and Chamberlain 1970).

No threatened or endangered plants or animals are known to be present in the area under consideration for development. Construction activities probably would not affect two small wetlands (Carolina bays) lying in the east-central portion of the undeveloped host site. Construction activities would not affect plant and animal diversity locally or regionally, because the managed loblolly and slash pine stands that would be removed are not unique, nor do they provide habitat for any protected, sensitive, unusual, or Federally listed plant or animal species.

Impacts of operations under this option would be similar to, but slightly greater than, those described for Option 2a. Overall impacts to ecological resources would be minor.

5.9.5.2 Option 5b - Wet Storage. Construction impacts under this option would be less than those described for Option 5a because less land area would be required for new facilities. Impacts of operations under this case would be similar to those described for Option 5a. Overall impacts to ecological resources would be minor.

5.9.5.3 Option 5c - Processing and Storage. Construction impacts under this case would be similar to those described for Option 5a. This case would require the largest number of workers of all the cases under consideration. It would result in more noise, more traffic, and a generally higher level of disturbance to terrestrial wildlife (specifically reptiles, songbirds, and small and large mammals) accustomed to feeding, foraging, perching, hunting, nesting, or denning in the area. Some animals would be driven from the area permanently, while others probably would become accustomed to the increased noise and activity levels, and would return to the area. Overall impacts to ecological resources would be minor.

5.9.5.4 Option 5d - Shipment off the Site. Construction impacts under this case would be smaller than those for any other alternative, excluding Alternative 1 - No Action. Impacts of operation under this case would also be minimal, limited to some minor disturbances of animals by vehicular traffic. Overall impacts to ecological resources would be minimal.

5.10 Noise

As described in Section 4.10, noises generated on the SRS do not travel off the Site at levels that affect the general population. Therefore, SRS noise impacts for each alternative would be limited to noise resulting from the transportation of personnel and materials to and from the Site that could affect nearby communities and from onsite sources that could affect some wildlife near these sources. DOE would address the effects of noise on wildlife near spent nuclear fuel management facilities under any alternative in a project-specific NEPA evaluation.

Transportation noises would be a function of the size of the workforce (i.e., an increased workforce would produce increased employee traffic and corresponding increases in deliveries by truck and rail and a decreased workforce would produce decreased employee traffic and corresponding decreases in deliveries). The analysis of traffic noise took into account railroad noise and noise from the major roadways that provide access to the SRS. DOE does not expect the number of freight trains per day in the region and through the Site to change as a result of any of the alternatives, although some trains could be dedicated to the transport of spent nuclear fuel. Rail shipments of spent nuclear fuel, regardless of the alternative, would not substantially increase the rail traffic on the CSX line through the SRS. Therefore, vehicles used to transport employees and personnel on roadways would be the principal sources of community noise impacts. This analysis used the day-night average sound level (DNL) to assess community noise, as suggested by the Environmental Protection Agency (EPA 1974; 1982) and the Federal Interagency Committee on Noise (FICON 1992). The analysis based its estimate of the change in day-night average sound level from the baseline noise level for each alternative on the projected changes in employment and traffic levels. The baseline levels are those for 1995. The analysis also considered the combination of construction and operation employment. The traffic noise analysis considered SC 125 and SC 19, both of which are used to access the SRS. Changes in noise level below 3 decibels would not be likely to result in a change in community reaction (FICON 1992).

DOE projects no new employment due to operations for any of the alternatives. Some additional construction jobs may be required but overall SRS employment would not exceed the 1995 baseline levels, except for Alternatives 5a, 5b, and 5c. The maximum Site employment of about 20,000 jobs would occur in 1995 for all alternatives except 5a, 5b, and 5c for which the peak would occur in about 2002 due to a peak in construction employment. The general decrease in employment after 1995 could result in some decrease in vehicle trips to and from the Site. There would be at most a few truck trips per day to and from the Site carrying spent nuclear fuel under any of the alternatives. This increase in truck trips would not result in a perceptible increase in traffic noise levels along the routes to the SRS. The day-night average sound level along SC 125 and SC 19 and other access routes would probably decrease slightly except in the peak construction years under Alternatives 5a, 5b, and 5c, as a result of the overall decrease in employment levels at the SRS after 1995. DOE expects no change in the community reaction to noise along these routes, and proposes no mitigation of traffic noise impacts.

5.11 Traffic and Transportation

This section discusses the consequences of both the onsite transportation of spent nuclear fuel and the increased traffic patterns due to construction activities at the SRS. Traffic due to operations of spent nuclear fuel facilities will remain at or below current Site levels because workers for the new activities will be drawn from the existing SRS workforce. The consequences of the transportation of spent fuel between the SRS and other DOE sites are described in Appendix I of this Environmental Impact Statement (EIS).

5.11.1 Traffic

Traffic impacts would be bound by Alternative 5b (Centralization - Wet Storage) which would result in the greatest number of additional construction workers (and vehicles) onsite. Level of service, a measure of traffic flow, was estimated for each road to and from the SRS. Traffic delays could be experienced at SC 19 and SC 230 intersections during peak hours. However, the number of construction vehicles in support of spent nuclear fuel construction activities would contribute less than 17 percent (HNUS 1994) to the total traffic flow. Therefore, the change in level of service due to Alternative 5b would be minimal.

5.11.2 Transportation

This section discusses the potential radiological consequences due to incident free transportation and accidents during transport. All SRS onsite shipments are carried out by rail.

5.11.2.1 Onsite Spent Nuclear Fuel Shipments. DOE based the number of fuel shipments on the amount and type of spent nuclear fuel stored at various SRS locations and the final storage location or disposition specified in the spent nuclear fuel alternatives. The number of shipments from each location was determined by dividing the amount of spent nuclear fuel at each location by the capacity of the shipping cask. Individual shipments from the various facilities were summed to obtain the total number of shipments for each alternative (HNUS 1994).

Onsite shipments are those that originate and terminate at the SRS. Movements of spent nuclear fuel within functional areas (e.g., H-Area or F-Area) are operational transfers, not onsite shipments; therefore, this analysis does not consider them.

5.11.2.2 Incident-Free Transportation Analysis. Under each alternative, DOE analyzed incident-free (normal transport) radiological impacts to transport vehicle crews and members of the general public from onsite rail shipments. The analysis calculated occupational radiation doses to the transport vehicle crew members (four locomotive operators). Because the general public does not have immediate access to areas where the SRS would transport spent nuclear fuel, the analysis assumed that any general public dose is to escorted individuals on the Site waiting at any of several train crossings at the time a fuel shipment passed. The analysis calculated radiological doses to the general public using the RISKIND (Yuan et al. 1993) computer code. The results are presented in Table 5-10.

The magnitude of incident-free consequence depends on the dose rate on the external surface of the transport vehicle, the exposure time, and the number of people exposed. For each receptor, the analysis assumed the external dose rate 2 meters (6.6 feet) from the shipping cask was 100 millirem per hour (HNUS 1994), which is the SRS procedurally-allowed maximum dose rate during onsite fuel shipments. Actual receptor dose rates would depend on receptor distance from the shipping cask [5 meters (16.4 feet) for the general public]. The duration of exposure would depend on the transport vehicle speed and the number of shipments. In addition, occupational exposure time would depend on the distance of each shipment.

The analysis calculated health effects measured as the number of latent cancer fatalities (LCFs) by multiplying the resultant occupational and general public doses by risk factors of 4×10^{-4} and 5×10^{-4} latent cancer fatalities per person-rem (DOE 1993a), respectively.

Table 5-10 summarizes the collective doses (person-rem) and health effects (latent cancer fatalities) associated with the incident-free onsite shipment of spent nuclear fuel at the SRS. Collective doses and latent cancer fatalities for members of the public would be approximately a factor of 10 less than those for the occupational worker. The data also indicate that the No-Action alternative would provide the least collective doses and least latent cancer fatalities.

5.11.2.3 Transportation Accident Analysis. DOE analyzed radiological impacts from potential accidents to both the onsite maximally exposed individual (MEI), and offsite members of the general public from onsite rail shipments. The analysis calculated doses using the RISKIND (Yuan et al. 1993) computer code with site-specific meteorology, demographics, and spent fuel activity. Risk was calculated using site-specific rail accident rates and accident probabilities (HNUS 1994).

The magnitude of accident consequence would depend on the amount of radioactive material to which the individual(s) was exposed, the exposure time, and the number of people exposed. The analysis assumed that the maximum reasonably foreseeable amount of radioactive material for the type of spent fuel shipped on the SRS was released (HNUS 1994). The assumed duration of exposure for each receptor was 2 hours. The assumed maximally exposed individual was an SRS worker downwind of the accident at distances of 50 and 100 meters (164 and 330 feet).

Table 5-10. Collective doses and health effects for onsite, incident-free spent nuclear fuel shipments by alternative.

Option	Occupational (person-rem)	General Public (person-rem)	Number of LCFs[a] Occupational	Number of LCFs[a] General Public
No Action				
Option 1b -Wet Storage	$1.5x10^{0}$	$1.4x10^{-1}$	$6.0x10^{-4}$	$7.0x10^{-5}$
Decentralization				
Option 2a - Dry Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 2b - Wet Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 2c - Processing	$5.3x10^{-1}$	$3.7x10^{-2}$	$2.1x10^{-4}$	$1.9x10^{-5}$
Planning Basis				
Option 3a - Dry Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 3b - Wet Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 3c - Processing	$5.3x10^{-1}$	$3.7x10^{-2}$	$2.1x10^{-4}$	$1.9x10^{-5}$
Regionalization				
Option 4a - Dry Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 4b - Wet Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 4c - Processing	$5.3x10^{-1}$	$3.7x10^{-2}$	$2.1x10^{-4}$	$1.9x10^{-5}$
Option 4d - Dry Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 4e - Wet Storage	$2.5x10^{-1}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 4f - Processing	$5.3x10^{-1}$	$3.7x10^{-2}$	$2.1x10^{-4}$	$1.9x10^{-5}$
Option 4g - Ship Out	NA[b]	NA[b]	NA[b]	NA[b]
Centralization				
Option 5a - Dry Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 5b - Wet Storage	$2.5x10^{0}$	$2.3x10^{-1}$	$1.0x10^{-3}$	$1.2x10^{-4}$
Option 5c - Processing	$5.3x10^{-1}$	$3.7x10^{-2}$	$2.1x10^{-4}$	$1.9x10^{-5}$
Option 5d - Ship Out	NA[b]	NA[b]	NA[b]	NA[b]

a. LCF = latent cancer fatality.
b. NA = not applicable.

The analysis calculated offsite exposure using both rural and suburban population density-specific census data. The rural and suburban population densities have an average of 6 persons per square kilometer and 244 persons per square kilometer, respectively. The west-northwest sector has the highest population density within 80 kilometers (50 miles) of the SRS.

The analysis used site-specific meteorology at the 50th and 95th percentile to determine dose consequences. Joint probability includes both the event frequency and the probability of the maximum reasonably foreseeable type of accident occurring.

The analysis calculated health effects measured as the number of latent cancer fatalities by multiplying the resultant occupational and general public doses by the risk factors of 4×10^{-4} and 5×10^{-4} latent cancer fatalities per person-rem (DOE 1993a), respectively. Risk was calculated by multiplying the resultant doses by the joint probability of 1×10^{-4} (HNUS 1994).

Tables 5-11 and 5-12 summarize the collective doses and associated latent cancer fatalities for postulated onsite rail accidents with subsequent releases of radioactive material to the environment. The dose consequences of an accidental release of radioactive material was assessed for the 95th and typical 50th percentile meteorological conditions (i.e., those that would result in lower doses 95 and 50 percent of the time, respectively). In all cases the estimated number of latent cancer fatalities would be low.

Table 5-11. Impacts on maximally exposed individual from spent nuclear fuel transportation accident on the Savannah River Site.

Dose Percentile	Distance (meters)	Dose to MEI[a] (rem)	Number of LCFs[b] per year	Risk
50 percent	100	0.16	6.4×10^{-5}	1.6×10^{-5}
95 percent	50	0.37	1.5×10^{-4}	3.7×10^{-5}

a. MEI = maximally exposed individual.
b. LCF = latent cancer fatality.

Table 5-12. Impacts on offsite population from spent nuclear fuel transportation accident on the Savannah River Site.

Population Density Category	Dose Percentile	Offsite Population Dose (person-rem)	Number of LCFs[a] per year	Risk
Rural	50th	1.7	8.7×10^{-4}	1.7×10^{-4}
Rural	95th	7.1	3.6×10^{-3}	3.6×10^{-3}
Suburban	50th	5.2	2.6×10^{-3}	2.6×10^{-3}
Suburban	95th	21.3	1.1×10^{-2}	1.1×10^{-2}

a. LCF = latent cancer fatality.

5.11.3 Onsite Mitigation and Preventative Measures

All onsite shipments must be in compliance with DOE Savannah River Directive Implementation Instruction 5480.3, "Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes." DOE, DOE-SR, or the Nuclear Regulatory Commission (NRC) must approve packages used for onsite shipments with a certificate of compliance. If DOE or NRC has not certified an onsite package as Type B, the shipper must establish administrative controls and site-mitigating circumstances that will ensure package integrity. The administrative and emergency response considerations must provide sufficient control so that accidents would not result in loss of containment, shielding, or criticality; or the uncontrolled release of radioactive material would not create a hazard to the health and safety of the public or workers.

In the event of an accident, SRS has established an emergency management program. This program incorporates activities associated with emergency planning, preparedness, and response.

5.12 Occupational and Public Health and Safety

5.12.1 Radiological Health

This human health effects analysis relied principally on data on F- and H-Area emissions documented for the 1985, 1986, and 1993 operating years (Marter 1986; 1987; WSRC 1994d). During the 1985-1986 period, F- and H-Areas processing facilities operated at high capacity; DOE believes, therefore, that these emissions represent conservative estimates as to the emissions that could result from spent nuclear fuel management activities at the SRS that involve processing. This air and surface-water emissions information defined the source terms for the baseline evaluation (No-Action alternative) of health effects discussed in this section. To estimate health effects, this analysis defined six human receptor groups:

- The F- and H-Area workers assigned to F- and H-Area operations involving nuclear materials
- The F- and H-Area workers assigned to the Receiving Basin for Offsite Fuels for storage operations.
- The maximally exposed individual residing at the SRS boundary
- The projected 1994 offsite population of 628,200 persons residing within an 80-kilometer (50-mile) radius of F- and H-Areas

- The maximally exposed individual potentially affected by SRS surface-water emissions

- The approximate offsite population of 65,000 persons whom SRS surface-water emissions could affect

With the exception of the worker group, this analysis calculated exposures for the remaining four receptor groups using the baseline source terms as input data to automated atmospheric and surface-water transport, human intake, and human dosimetry models configured for routine use at SRS (Hamby 1994). The analysis estimated worker exposures using averaged dosimetry data recorded for F- and H-Area workers from 1983 through 1987 and Receiving Basin for Offsite Fuels workers for 1993 (Matheny 1994), corrected for an assumed occupancy factor of 0.25 (i.e., a worker could be potentially exposed during one-quarter of his/her shift). This correction was applied to the 1983-1987 data only. At the SRS, the waterborne exposure pathway does not exist for the worker receptor group because Site drinking water is drawn from deep aquifers unaffected by any radiological releases.

The analysis developed incremental receptor group exposure estimates (millirem per year, person-rem per year; effective dose equivalent) based on spent fuel quantities for each of the nonbaseline alternatives (i.e., Alternatives 2 through 5) and their options by applying calculated ratios of metric tons of heavy metal (MTHM) for each alternative and option compared to the No-Action alternative. DOE used these ratios as incremental scaling factors to estimate exposures under each option. The calculation of the MTHM ratios used the data presented in Table 3.1. Table 5-13 lists the results of the exposure estimate calculations. Since these incremental exposures include contributions to the effective dose equivalent from existing (No Action) spent fuel management at the SRS, the change in health effects for each alternative can be estimated as the difference between the alternatives presented.

The analysis calculated the potential health effects expressed in the exposed receptor groups consistent with risk determination guidance issued by the DOE Office of NEPA Oversight (DOE 1993a) and International Commission on Radiological Protection Publication 60 (ICRP 1991). For exposed individuals and populations, the potential health effect (detriment) of interest is latent fatal cancer. For exposed individuals, this analysis presents the health effect as the maximum incremental probability for detriment expression; for exposed populations, it presents the annual incremental detriment incidence. For completeness, it also provides the "project life" (i.e., 40 years) detriment incidence as the annual incidence multiplied by 40. Table 5-14 (worker) and Table 5-15 (maximally exposed individual and offsite population) summarize the health effects calculations.

Table 5-13. Incremental radioactive contaminant annual exposure summary.

Alternative	Onsite Workers[a] (mrem/year)	Onsite Workers[a] (person-rem/year)	MEI Offsite[a,b,d] (mrem/year) Air	MEI Offsite[a,b,d] (mrem/year) Water	Offsite Population[a,d] (person-rem/year Air	Offsite Population[a,d] (person-rem/year Water
No Action - Wet Storage (Option 1)	100	0.2	9×10^{-8}	3×10^{-8}	4×10^{-6}	6×10^{-7}
Decentralization - Drv Storage (Option 2a)	83	0.2	8×10^{-8}	2×10^{-8}	3×10^{-6}	5×10^{-7}
Decentralization - Wet Storage (Option 2b)	104	0.2	9×10^{-8}	3×10^{-8}	4×10^{-6}	6×10^{-7}
Decentralization - Processing (Option 2c)	145	70	0.4	0.1	14	2.2
Planning Basis - Dry Storage (Option 3a)	84	0.2	8×10^{-8}	2×10^{-8}	3×10^{-6}	5×10^{-7}
Planning Basis - Wet Storage (Option 3b)	105	0.2	1×10^{-7}	3×10^{-8}	4×10^{-6}	6×10^{-7}
Planning Basis - Processing (Option 3c)	147	71	0.4	0.1	15	2.2
Regionalization A - Dry Storage (Option 4a)	83	0.2	8×10^{-8}	2×10^{-8}	3×10^{-6}	5×10^{-7}
Regionalization A - Wet Storage (Option 4b)	103	0.2	9×10^{-8}	3×10^{-8}	4×10^{-6}	6×10^{-7}
Regionalization A - Processing (Option 4c)	148	76	0.4	0.1	16	2.4
Regionalization B - Dry Storage (Option 4d)	105	0.2	1×10^{-7}	3×10^{-8}	4×10^{-6}	6×10^{-7}
Regionalization B - Wet Storage (Option 4e)	131	0.3	1×10^{-7}	4×10^{-8}	5×10^{-6}	7×10^{-7}
Regionalization B - Processing (Option 4f)	175	74	0.4	0.1	15	2.3
Regionalization B - Ship Out (Option 4g)	<100	<0.2	$<9 \times 10^{-8}$	$<3 \times 10^{-8}$	$<4 \times 10^{-6}$	$<6 \times 10^{-7}$
Centralization - Dry Storage (Option 5a)	1,102[c]	2.2	1×10^{-6}	3×10^{-7}	4×10^{-5}	6×10^{-6}
Centralization - Wet Storage (Option 5b)	1,377[c]	2.8	1×10^{-6}	4×10^{-7}	5×10^{-5}	8×10^{-6}
Centralization - Processing (Option 5c)	1,422[c]	79	0.4	0.1	16	2.4
Centralization - Ship Out (Option 5d)	<100	<0.2	$<9 \times 10^{-8}$	$<3 \times 10^{-8}$	$<4 \times 10^{-6}$	$<6 \times 10^{-7}$

a. Insignificant digits are displayed for comparison purposes only.
b. MEI = maximally exposed individual.
c. The DOE regulatory exposure limit is 2,000 mrem (DOE 1992).
d. Data is provided separately for the air and water exposure pathways because the receptors are not co-located.

Table 5-14. Incremental fatal cancer incidence and maximum probability for workers.

Alternative	Annual Incidence[a]	40-Year Incidence	Maximum Probability
No Action - Wet Storage (Option 1)	$8x10^{-5}$	$3x10^{-3}$	$4x10^{-5}$
Decentralization - Dry Storage (Option 2a)	$7x10^{-5}$	$3x10^{-3}$	$3x10^{-5}$
Decentralization - Wet Storage (Option 2b)	$8x10^{-5}$	$3x10^{-3}$	$4x10^{-5}$
Decentralization - Processing (Option 2c)	$3x10^{-2}$	1	$6x10^{-5}$
Planning Basis - Dry Storage (Option 3a)	$7x10^{-5}$	$3x10^{-3}$	$3x10^{-5}$
Planning Basis - Wet Storage (Option 3b)	$8x10^{-5}$	$3x10^{-3}$	$4x10^{-5}$
Planning Basis - Processing (Option 3c)	$3x10^{-2}$	1	$6x10^{-5}$
Regionalization A - Dry Storage (Option 4a)	$7x10^{-5}$	$3x10^{-3}$	$3x10^{-5}$
Regionalization A - Wet Storage (Option 4b)	$8x10^{-5}$	$3x10^{-3}$	$4x10^{-5}$
Regionalization A - Processing (Option 4c)	$3x10^{-2}$	1	$6x10^{-5}$
Regionalization B - Dry Storage (Option 4d)	$8x10^{-5}$	$3x10^{-3}$	$4x10^{-5}$
Regionalization B - Wet Storage (Option 4e)	$1x10^{-4}$	$4x10^{-3}$	$5x10^{-5}$
Regionalization B - Processing (Option 4f)	$3x10^{-2}$	1	$7x10^{-5}$
Regionalization B - Ship Out (Option 4g)	$<8x10^{-5}$	$<3x10^{-3}$	$<4x10^{-5}$
Centralization - Dry Storage (Option 5a)	$9x10^{-4}$	$4x10^{-2}$	$4x10^{-4}$
Centralization - Wet Storage (Option 5b)	$1x10^{-3}$	$4x10^{-2}$	$5x10^{-4}$
Centralization - Processing (Option 5c)	$3x10^{-2}$	1	$6x10^{-4}$
Centralization - Ship Out (Option 5d)	$<8x10^{-5}$	$<3x10^{-3}$	$<4x10^{-5}$

a. Number of latent fatal cancers over a lifetime which could be attributed to one year of spent nuclear fuel management activities.

The Centers for Disease Control and Prevention is conducting a comprehensive reconstruction of historic offsite doses associated with SRS operations. The results of this investigation are not yet available.

5.12.2 Nonradiological Health

DOE used the operations air quality data listed in Tables 5-3, 5-4, 5-5 and 5-6 (and Table 8 of WSRC 1994a) to evaluate health impacts associated with potential exposure to the following two compound classes: criteria pollutants and toxic pollutants. The analysis evaluated two hypothetical receptor locations: (1) a worker in S-Area and (2) a maximally exposed individual at the SRS boundary. However, it was unnecessary to postulate an intake of criteria pollutant or toxic compounds by these receptors because airborne concentration standards are available for these compounds.

Tables 5-3 and 5-4 lists 8 criteria pollutants and 23 toxic compounds. The toxic compounds were classified as carcinogens and noncarcinogens consistent with Environmental Protection Agency carcinogenicity group (weight of evidence) designations published in the Integrated Risk Information

Table 5-15. Incremental fatal cancer incidence and maximum probability for the maximally exposed individual and offsite population (air and water pathways).

Alternative	Population Annual Incidence[a]	Population 40-Year Incidence	MEI Maximum Probability
No Action - Wet Storage (Option 1)			
Air	2x10^{-9}	7x10^{-8}	4x10^{-14}
Water	3x10^{-10}	1x10^{-8}	1x10^{-14}
Decentralization - Dry Storage (Option 2a)			
Air	2x10^{-9}	6x10^{-8}	4x10^{-14}
Water	2x10^{-10}	9x10^{-9}	1x10^{-14}
Decentralization - Wet Storage (Option 2b)			
Air	2x10^{-9}	8x10^{-8}	5x10^{-14}
Water	3x10^{-10}	1x10^{-8}	2x10^{-14}
Decentralization - Processing (Option 2c)			
Air	7x10^{-3}	0.3	2x10^{-7}
Water	1x10^{-3}	4x10^{-2}	6x10^{-8}
Planning Basis - Dry Storage (Option 3a)			
Air	2x10^{-9}	6x10^{-8}	4x10^{-14}
Water	2x10^{-10}	9x10^{-9}	1x10^{-14}
Planning Basis - Wet Storage (Option 3b)			
Air	2x10^{-9}	8x10^{-8}	5x10^{-14}
Water	3x10^{-10}	1x10^{-8}	2x10^{-14}
Planning Basis - Processing (Option 3c)			
Air	7x10^{-3}	0.3	2x10^{-7}
Water	1x10^{-3}	4x10^{-2}	6x10^{-8}
Regionalization A - Dry Storage (Option 4a)			
Air	2x10^{-9}	6x10^{-8}	4x10^{-14}
Water	2x10^{-10}	9x10^{-9}	1x10^{-14}
Regionalization A - Wet Storage (Option 4b)			
Air	2x10^{-9}	8x10^{-8}	5x10^{-14}
Water	3x10^{-10}	1x10^{-8}	2x10^{-14}
Regionalization A - Processing (Option 4c)			
Air	8x10^{-3}	0.3	2x10^{-7}
Water	1x10^{-3}	5x10^{-2}	6x10^{-8}
Regionalization B - Dry Storage (Option 4d)			
Air	2x10^{-9}	8x10^{-8}	5x10^{-14}
Water	3x10^{-10}	1x10^{-8}	2x10^{-14}
Regionalization B - Wet Storage (Option 4e)			
Air	2x10^{-9}	1x10^{-7}	6x10^{-14}
Water	4x10^{-10}	1x10^{-8}	2x10^{-14}
Regionalization B - Processing (Option 4f)			
Air	8x10^{-3}	0.3	2x10^{-7}
Water	1x10^{-3}	5x10^{-2}	6x10^{-8}
Regionalization B - Ship Out (Option 4g)			
Air	<2x10^{-9}	<7x10^{-8}	<4x10^{-14}
Water	<3x10^{-10}	<1x10^{-8}	<1x10^{-14}

Table 5-15. (continued).

Alternative	Population Annual Incidence[a]	Population 40-Year Incidence	MEI Maximum Probability
Centralization - Dry Storage (Option 5a)			
Air	2×10^{-8}	8×10^{-7}	5×10^{-13}
Water	3×10^{-9}	1×10^{-7}	2×10^{-13}
Centralization - Wet Storage (Option 5b)			
Air	3×10^{-8}	1×10^{-6}	6×10^{-13}
Water	4×10^{-9}	2×10^{-7}	2×10^{-13}
Centralization - Processing (Option 5c)			
Air	8×10^{-3}	0.3	2×10^{-7}
Water	1×10^{-3}	5×10^{-2}	6×10^{-8}
Centralization - Ship Out (Option 5d)			
Air	$<2\times10^{-9}$	$<7\times10^{-8}$	$<4\times10^{-14}$
Water	$<3\times10^{-10}$	$<1\times10^{-8}$	$<1\times10^{-14}$

a. Number of latent fatal cancers over a lifetime that could be attributed to one year of spent nuclear fuel management activities.

System (IRIS) data base (DOE 1994). For purposes of health effects analysis, carcinogens are those compounds designated Group A (human carcinogens), Group B1 (probable human carcinogen, limited evidence in human studies), Group B2 (probable human carcinogen, inadequate evidence or no data from human studies), and Group C (possible human carcinogen). Using this designation, three of the 23 toxic compounds are carcinogens: benzene (Group A), formaldehyde (Group B1), and methylene chloride (Group B2).

Carcinogen health effects are expressed as the incremental probability of an individual developing cancer, assuming a lifetime (70 years) of exposure to the carcinogen. DOE used cancer risk (slope) factors published in IRIS (Integrated Risk Information System) to obtain unit risk factors (risk per concentration) needed to calculate incremental probability. Carcinogens with insufficient (i.e., incomplete or unavailable carcinogen assessment data) information listed in the Integrated Risk Information System data base precluded a quantitative risk assessment; this analysis evaluated them as noncarcinogens.

This analysis evaluated noncarcinogenic and priority pollutant compound health effects by adding hazard quotients to obtain a hazard index. The hazard quotient is the ratio of compound concentration or dose to a Reference Concentration (RfC) or Dose (RfD) (EPA 1989). The regulatory standard used in this analysis was the more stringent of the following: (1) Occupational Safety and Health Administration (OSHA) 8-hour permissible exposure limit (PEL), (2) American Conference of

Governmental Industrial Hygienists (ACGIH) threshold limit value (TLV), or (3) State of South Carolina air quality standards. The use of the noncancer hazard index assumed a level of exposure (i.e., RfC) below which adverse health effects are unlikely. The hazard index is not a statistical probability; therefore it cannot be interpreted as such.

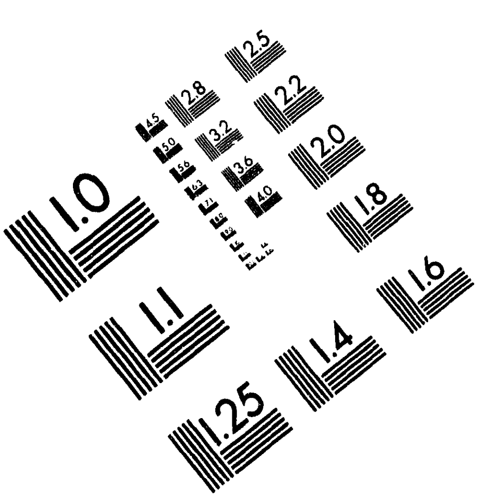
Table 5-16 summarizes nonradiological health effects attributable to atmospheric emissions of toxic and criteria pollutant compounds. Because no hazard index value would exceed unity (1.0), adverse health effects are unlikely under any alternative.

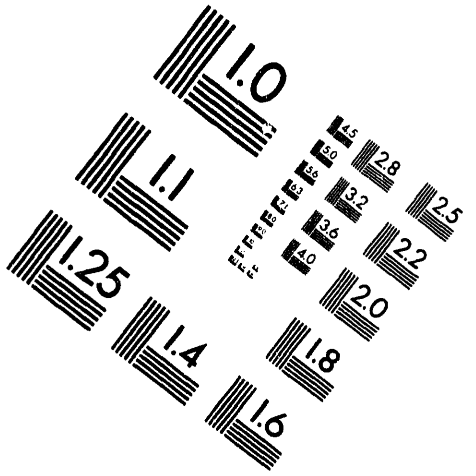
5.12.3 Industrial Safety

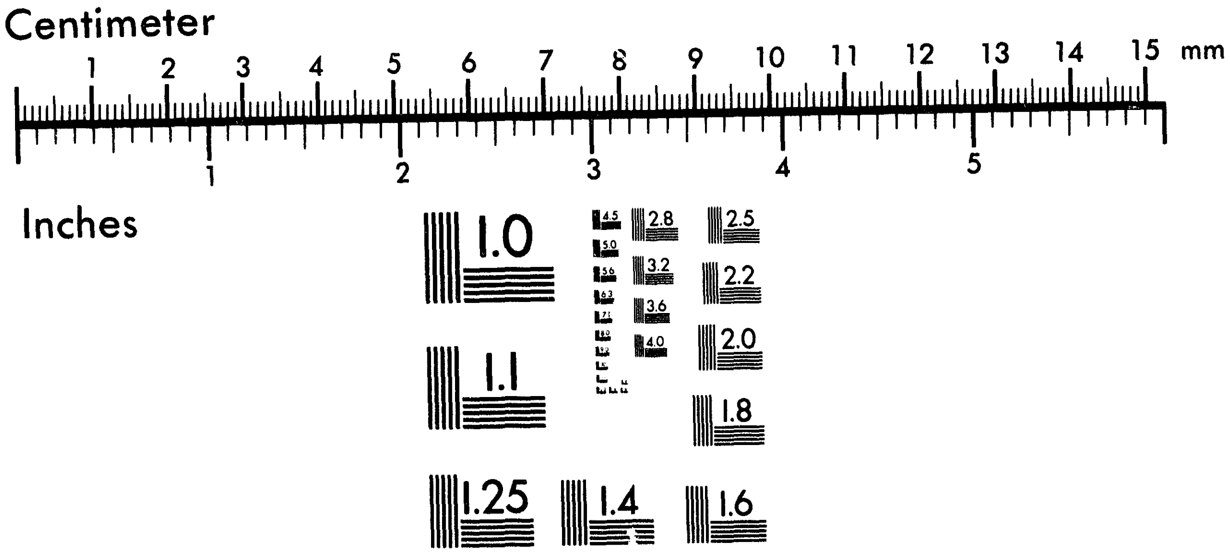
This section describes the following measures of impact for workplace hazards: (1) total reportable injuries and illnesses and (2) fatalities in the work force. This analysis considers injury/illness and fatality incidence rates for construction workers separately because of the relatively more hazardous nature of construction work. Table 5-16 lists the incidence of injuries/illnesses and fatalities for construction and non-construction workers. These data are for the highest employment year (i.e., maximum hours worked in any year from 1994 through 2035, assuming 2,000 hours per worker) (WSRC 1994). This analysis used the average occupational injury/illness and fatality incidence rates experienced by DOE and its contractors from 1988 through 1992 to calculate the incidence of industrial hazards listed in Table 5-17 (DOE 1993b).

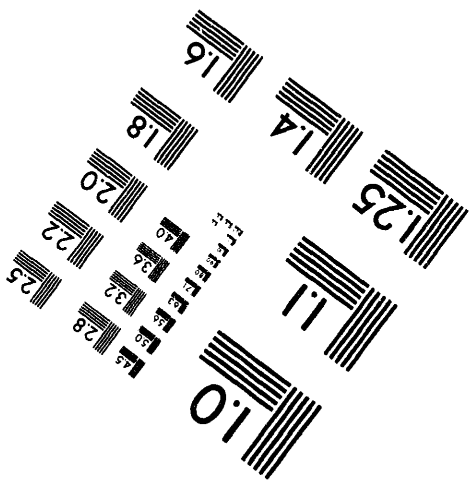
5.13 Utilities and Energy

The existing capacities and distribution systems at the SRS for electricity, steam, water, and domestic wastewater treatment are adequate to support any of the five alternatives. Table 5-18 summarizes estimates of the annual requirements for electricity, steam, and domestic wastewater treatment for each alternative and case, and compares them to current SRS usage of these resources. Table 5-8 lists information on water usage by alternative. The utility and energy requirements for all the alternatives represent a small percentage of current requirements. No new generation or treatment facilities would be necessary; connections to existing networks would require only short tie-in lines. Increases in SRS fuel consumption would be minimal because overall activity on the Site would not increase due to changes in the SRS mission and the general reduction in employment levels. The overall impacts of any of the alternatives on the SRS utilities and energy resources would be minimal.









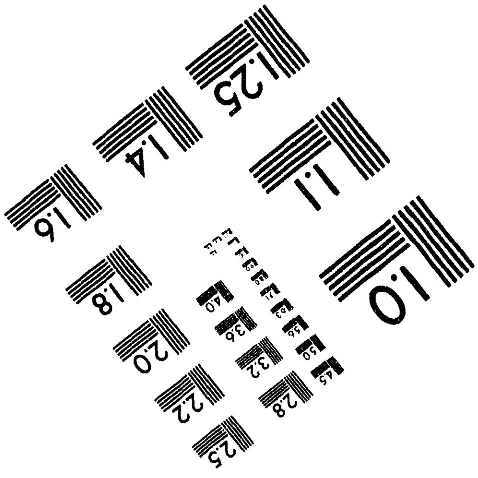


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Table 5-16. Nonradiological annual incremental health effects summary.[a]

Alternative	Worker Cancer Probability[b]	Worker Hazard Index	MEI Cancer Probability[b,c]	MEI Hazard Index
No Action - Wet Storage (Option 1)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Decentralization - Dry Storage (Option 2a)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Decentralization - Wet Storage (Option 2b)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Decentralization - Processing (Option 2c)	Insufficient data	$6x10^{-3}$	Insufficient data	$5x10^{-4}$
Planning Basis - Dry Storage (Option 3a)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Planning Basis - Wet Storage (Option 3b)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Planning Basis - Processing (Option 3c)	Insufficient data	$6x10^{-3}$	Insufficient data	$5x10^{-4}$
Regionalization A - Dry Storage (Option 4a)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Regionalization A - Wet Storage (Option 4b)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Regionalization A - Processing (Option 4c)	Insufficient data	$6x10^{-3}$	Insufficient data	$5x10^{-4}$
Regionalization B - Dry Storage (Option 4d)	Insufficient data	$2x10^{-6}$	Insufficient data	$3x10^{-7}$
Regionalization B - Wet Storage (Option 4e)	Insufficient data	$2x10^{-6}$	Insufficient data	$3x10^{-7}$
Regionalization B - Processing (Option 4f)	Insufficient data	$8x10^{-3}$	Insufficient data	$6x10^{-4}$
Regionalization B - Ship Out (Option 4g)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Centralization - Dry Storage (Option 5a)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Centralization - Wet Storage (Option 5b)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$
Centralization - Processing (Option 5c)	Insufficient data	$6x10^{-3}$	Insufficient data	$5x10^{-4}$
Centralization - Ship Out (Option 5d)	Insufficient data	$2x10^{-6}$	Insufficient data	$2x10^{-7}$

a. Source: DOE (1991).
b. Insufficient data exists in the IRIS data base to perform a quantitative inhalation cancer risk assessment.
c. MEI = maximally exposed individual.

Table 5-17. Incremental industrial hazard maximum annual incidence summary.

Alternative	Construction Injuries and Illnesses	Construction Fatalities	Nonconstruction Injuries and Illnesses	Nonconstruction Fatalities
No Action - Wet Storage (Option 1)	92	<1	159	<1
Decentralization - Dry Storage (Option 2a)	71	<1	159	<1
Decentralization - Wet Storage (Option 2b)	71	<1	159	<1
Decentralization - Processing (Option 2c)	66	<1	159	<1
Planning Basis - Dry Storage (Option 3a)	71	<1	159	<1
Planning Basis - Wet Storage (Option 3b)	82	<1	159	<1
Planning Basis - Processing (Option 3c)	66	<1	159	<1
Regionalization A - Dry Storage (Option 4a)	82	<1	159	<1
Regionalization A - Wet Storage (Option 4b)	82	<1	159	<1
Regionalization A - Processing (Option 4c)	66	<1	159	<1
Regionalization B - Dry Storage (Option 4d)	89	<1	199	<1
Regionalization B - Wet Storage (Option 4e)	102	<1	199	<1
Regionalization D - Processing (Option 4f)	82	<1	199	<1
Regionalization B - Ship Out (Option 4g)	22	<1	159	<1
Centralization - Dry Storage (Option 5a)	316	1	159	<1
Centralization - Wet Storage (Option 5b)	337	1	159	<1
Centralization - Processing (Option 5c)	316	1	159	<1
Centralization - Ship Out (Option 5d)	22	<1	159	<1

The smallest increase in demand would result from the No-Action alternative, which would be similar to current spent nuclear fuel-related requirements at the SRS. The largest increases would be due to the centralization of spent nuclear fuel at the SRS (Alternative 5). Alternative 5 would result in a maximum additional electrical demand of about 110,400 megawatt-hours annually (Option 5c), and an increased steam consumption of about 19.1 million kilograms (42.1 million pounds) per year (Option 5c). Water requirements would also be greatest under this Alternative (Table 5-6). Annual withdrawals of Savannah River water for cooling purposes would reach about 310.8 million liters (82.1 million gallons) and groundwater usage for domestic and processing purposes would total approximately 69.6 million liters (18.4 million gallons). The volume of domestic wastewater requiring treatment would range from approximately 35 to 70 million liters (9 to 18 million gallons) per year. This additional water usage amounts to an increase of about 10 percent over current SRS water requirements.

Among the three management options, processing would result in the greatest increase in demand on utilities and energy in comparison to either the dry or wet storage options. In general, dry and wet storage would be similar in their requirements of these resources.

5.14 Materials and Waste Management

This section discusses potential impacts of the management of materials and wastes associated with the implementation of alternatives identified for spent nuclear fuel management. Sections 5.7 and 5.12 (Air Quality and Occupational and Public Health and Safety, respectively) discuss the impacts of hazardous and toxic materials as they relate to routine operations and accidents.

DOE has projected rates and volumes of waste and impacts of waste generation at SRS for low-level, transuranic, and high-level wastes for each of the alternatives for spent nuclear fuel management. Table 5-19 summarizes the estimated annual average and total volume of these three waste types that each alternative would produce during a 40-year management period. The discussion below also identifies the impacts that the waste produced by spent nuclear fuel activities would have on the existing SRS capacity to manage each waste type.

Table 5-18. Estimates of annual electricity, steam, and domestic wastewater treatment requirements for each alternative.[a,b]

Alternative	Electricity Usage (megawatt hours per year)	Steam Usage (kilograms per year)[c]	Domestic Wastewater Treatment (liters per year)[d]
Current SRS Usage	659,000	1.7 billion	690 million
1. No Action			
Option 1 - Wet Storage	1,400	11.3 million	35.1 million
2. Decentralization			
Option 2a - Dry Storage	19,400	16.7 million	48.7 million
Option 2b - Wet Storage	22,400	14.4 million	50.6 million
Option 2c - Processing	56,400	19.1 million	48.7 million
3. 1992/1993 Planning Basis			
Option 3a - Dry Storage	19,400	16.7 million	48.7 million
Option 3b - Wet Storage	22,400	14.4 million	50.6 million
Option 3c - Processing	56,400	19.1 million	48.7 million
4. Regionalization - A			
Option 4a - Dry Storage	24,400	16.7 million	48.7 million
Option 4b - Wet Storage	27,400	14.4 million	50.6 million
Option 4c - Processing	67,400	16.5 million	47.6 million
Regionalization - B			
Option 4d - Dry Storage	24,400	16.7 million	48.7 million
Option 4e - Wet Storage	27,400	14.4 million	50.6 million
Option 4f - Processing	56,400	19.1 million	48.7 million
Option 4g - Ship Out	11,400	11.7 million	38.1 million
5. Centralization			
Option 5a - Dry Storage	44,400	16.7 million	67.7 million
Option 5b - Wet Storage	47,400	14.4 million	69.6 million
Option 5c - Processing	110,400	19.1 million	67.7 million
Option 5d - Ship Out	11,400	11.7 million	38.1 million

a. Source: WSRC (1994b).
b. Water requirements are shown in Table 5-8.
c. To convert kilograms to pounds, multiply by 2.2046.
d. To convert liters to gallons, multiply by 0.26418.

Table 5-19. Annual average and total volume (cubic meters)[d] of radioactive wastes produced under each alternative during the 40-year interim management period.[a]

Alternative	Low-level waste[b]		Transuranic waste		High-level waste[c]	
	Average	Total	Average	Total	Average	Total
1. No Action						
Option 1 - Wet Storage	400	17,600	17	700	0	0
2. Decentralization						
Option 2a - Dry Storage	400	17,600	18	730	0	0
Option 2b - Wet Storage	400	17,600	18	730	0	0
Option 2c - Processing	1,700	68,000	20	780	2	19
3. 1992/1993 Planning Basis						
Option 3a - Dry Storage	400	17,600	18	730	0	0
Option 3b - Wet Storage	400	17,600	18	730	0	0
Option 3c - Processing	1,700	68,000	20	780	2	19
4. Regionalization - A						
Option 4a - Dry Storage	400	17,600	18	730	0	0
Option 4b - Wet Storage	400	17,600	18	730	0	0
Option 4c - Processing	1,700	68,000	20	780	2	19
4. Regionalization - B						
Option 4d - Dry Storage	400	17,600	18	730	0	0
Option 4e - Wet Storage	400	17,600	18	730	0	0
Option 4f - Processing	1,700	68,000	20	780	2	19
Option 4g - Ship Out	400	16,300	5	180	0	0
5. Centralization						
Option 5a - Dry Storage	400	17,600	18	730	0	0
Option 5b - Wet Storage	500	20,000	20	780	0	0
Option 5c - Processing	1,700	68,000	20	780	2	19
Option 5d - Ship Out	400	16,300	5	180	0	0

a. Based on WSRC (1994b).
b. Source: WSRC (1994c).
c. Figures are for the initial 10-year period when most processing would be completed.
d. To convert cubic meters to cubic yards multiply by 1.307.

DOE has not developed estimates of low-level mixed, hazardous, or solid sanitary wastes that spent nuclear fuel management activities at the SRS could generate, although it is anticipated that these activities would produce these waste types only in limited quantities. Further, the discussions in Section 5.14.2 related to the impacts of spent fuel management wastes on the SRS waste capacities do not include considerations of wastes that will result from Site cleanup because assessments for these activities are still underway and will undergo National Environmental Policy Act review as part of the SRS Waste Management Environmental Impact Statement (59 FR 16194; 4/6/94).

Volume 1 of this spent nuclear fuel EIS provides information concerning the major Federal environmental laws and regulations, Executive Orders, and DOE Orders that apply to pollution prevention at the Savannah River Site. The DOE views source reduction as the first priority in its pollution prevention program, followed by an increased emphasis on recycling. Source reduction will reduce the waste management burden while eliminating the potential for future liability and cleanup. Recycling and using recycled materials will conserve resources and landfill space. Waste treatment and disposal are considered only when prevention or recycling is not possible or practical. Since creating a Savannah River Site waste minimization program (the precursor of the SRS pollution prevention program) in 1990, the amounts of wastes of all types (excluding low-level wastes, which are a by-product of environmental restoration activities) generated have decreased, with greatest reductions in hazardous and mixed wastes (Hoganson and Miles 1994).

5.14.1 Alternative Comparison

The first four alternatives would generate similar amounts of radioactive waste because the activities that produce the wastes would be similar under each of the alternatives. Most of the low-level and transuranic wastes would be generated during the first part of the 40-year management period while DOE was transferring existing inventory and renovating the Receiving Basin for Offsite Fuels and a reactor basin. The characterization and canning of the current inventory prior to placement into storage would also result in some waste generation. Once in storage, management activities would produce only small amounts of radioactive waste for the rest of the 40-year period.

The dry- and wet-storage options would both produce about 17,600 cubic meters (23,003 cubic yards) of low-level waste and about 730 cubic meters (954 cubic yards) of transuranic waste during the 40-year management period. Neither option would generate any high-level waste. The processing of the existing aluminum-clad fuels and storage of the others (the third option under each alternative) would generate all three types of waste: low-level and high-level wastes in appreciably greater volumes, and transuranic waste in slightly-greater volumes (780 cubic meters over 40 years versus 730 cubic meters for storage options).

Alternative 5 (for those options where DOE centralizes the spent nuclear fuel at the SRS) would result in somewhat larger volumes of radioactive waste than the other four alternatives. The increase in waste would not be directly proportional to the larger amounts of fuel that would be managed on the Site, because most of the originating sites would characterize and can their fuel prior to shipment so that it could be placed directly into storage at the SRS. Therefore, the radioactive wastes produced during centralization at the Site would come from the initial fuel transfer and pool renovations and from characterizing and canning small amounts of new fuel. The wet storage option would generate more waste than dry storage under this alternative because of maintenance of the pools required to store the large inventory. The processing of existing aluminum-clad fuels would produce the same types and volumes of waste as for the other alternatives.

The option for shipping the SRS inventory off the Site for regionalization or centralization elsewhere would also result in the production of some radioactive waste. This would occur during characterization and canning prior to shipment and would generate the smallest volumes of waste of any alternative action: 16,300 cubic meters (21,304 cubic yards) of low-level waste and 180 cubic meters (235 cubic yards) of transuranic waste. This waste would be produced only during the initial 10 years of the management period.

5.14.2 Impact on the SRS Waste Management Capacity

The impact of spent nuclear fuel activities on SRS waste management capacities would be minimal because the Site could accommodate the waste with existing and planned radioactive waste storage and disposal facilities. DOE would transfer high-level waste to the F/H Tank Farms for volume reduction and then to the Defense Waste Processing Facility (DWPF) for conversion into a borosilicate glass form suitable for prolonged storage. The SRS would use the Consolidated Incineration Facility, once operational, to treat the low-level waste. This facility has sufficient permitted capacity [105,500 cubic meters (137,889 cubic yards) per year] to treat the anticipated volume of these materials. However, actual through-put volume is dependent upon operational variables and waste characteristics. The F/H Effluent Treatment Facility would treat liquid low-level waste. This facility has sufficient design process capacity [598 million liters (158 million gallons) per year] to treat the anticipated volumes of these materials. DOE would manage the transuranic wastes with existing and planned storage capacity.

5.15 Accident Analysis

Operations involving the receipt, handling, processing, or storing of spent nuclear fuel would involve radioactive materials or toxic chemicals. These materials would be received, treated, stored, transferred between facilities, disposed of on the Site, and shipped off the Site. Under certain circumstances, these materials could be involved in an accident.

An accident is a series of unexpected or undesirable events initiated by equipment failure, human error, or a natural phenomenon such as severe weather, earthquake, or volcanism. These events can cause the release of either radioactive or chemically toxic materials inside a facility or to the environment.

This section summarizes analyses of possible accidents involving spent nuclear fuel operations at the SRS. To provide a perspective on potential accidents, this section summarizes various accidents associated with spent nuclear fuel activities that have occurred at the SRS (historic accidents) and reviews previous accident analyses for Site operations. This section uses the results of previous analyses as a baseline for determining the impacts for the alternatives that involve new facilities. For each alternative, this section discusses the accidents with the largest point estimates of risk (radiological impacts in terms of potential fatal cancers x frequency of the initiating event).

The facilities considered for each alternative are either existing facilities for which the approved safety analyses were used, or new facilities (WSRC 1994b) for which existing safety analysis results were substituted by evaluating the type of accident(s) that could be postulated to occur based on the projected function of the facility. Two facilities that contain very small amounts of spent nuclear fuel, Buildings 331-M and 773-A, were not included in this analysis because accidents analyzed for the major facilities would bound the consequences of possible accidents in these two locations.

This section addresses historic accidents, facility radiological accidents, chemical hazard accidents, and secondary impacts. Section 5.11 addresses onsite transportation accidents.

5.15.1 Historic Accidents at the Savannah River Site

Impacts from accidents can involve fatalities, injuries, or illness. Fatalities can be prompt (immediate) such as in construction accidents or latent (delayed) such as an increase in latent fatal cancers due to radiation exposure. Section 5.12 addresses worker injuries, illnesses, and the potential for increased cancer risk anticipated from normal operations of the facilities. Nonradiation accidents have dominated impacts to workers at the SRS (Durant 1987); impacts to the public from historic SRS accidents have been negligible.

The SRS has maintained an operational event data base on its facilities since the 1950s. This data base currently contains approximately 450,000 entries including data on the Receiving Basin for Offsite Fuel, the principal wet storage pool facility at the SRS; and both F-and H-Area Canyons. For this EIS, DOE reviewed the data base to identify historic spent nuclear fuel-related accidents at these facilities. Fuel cutting events, fuel handling events, and various liquid releases related to spent nuclear fuel management over the 40-year operating history of the SRS were examined. The purpose of the data base review was to provide an historic perspective on the types of accidents that have occurred at the SRS. Events representative of fuel failures include higher than expected contamination levels in fuel storage basin water and evidence of fuel canister cracking at a weld. Fuel handling incidents were due in large part to crane operator errors or crane and handling equipment failures. The data base also includes reports of incorrect fuel cropping, where the active region of fuel was exposed under water. These historical events provided a basis for the selection of representative accidents covering the spectrum of spent nuclear fuel management activities. No significant offsite impacts have resulted from these historic occurrences.

5.15.2 Potential Facility Accidents

The SRS spent nuclear fuel alternatives have the potential for radiological accidents (see Attachment A, Table A-2) that could affect the health and safety of workers and the public. The concerns and characteristics that are common to these accidents would be common regardless of whether the cause were a natural phenomenon or human error. For health effects to occur, an accident must allow a release of hazardous material to, or an increase in radiation levels in, the facility or the environment. The released material must be transported to locations frequented by humans. The quantities of hazardous materials that reach locations where people are and the ways they interact with people are important factors in the determination of health effects.

A number of studies have investigated the ways in which radioactivity reaches humans, how the body absorbs and retains it, and the resulting health effects. The International Commission on Radiological Protection has made specific recommendations for estimating these health effects (ICRP 1991). This organization is the recognized body for establishing standards for the protection of workers and the public from the effects of radiation exposure. Health effects include acute damage (up to and including death) and latent effects, including cancers and genetic damage. An SRS-developed computer code, AXAIR89Q, estimates potential radiation doses to maximally exposed individuals or population groups from accidental releases of radionuclides.

The AXAIR89Q code is a highly automated site-specific environmental dispersion and dosimetry code for postulated airborne releases. The environmental dispersion models used are based on NRC

Regulatory Guide 1.145 (NRC 1983). The exposure pathways considered in the AXAIR89Q code include inhalation of radionuclides and gamma irradiation from the radioactive plume.

Doses from the inhalation of radionuclides in air depend on the amount of radionuclides released; the dispersion factor; the physical, chemical, and radiological characteristics of the radionuclides; and various biological parameters such as breathing rate and biological half-life. The AXAIR89Q code uses a conservative breathing rate of 12,000 cubic meters (424,000 cubic feet) per year for adults. The dose commitment factors used in the environmental dosimetry code, as described in the following section, are from the inhalation dose conversion factors in International Committee on Radiation Protection Publication No. 30 (ICRP-30).

External gamma radiation doses from the traveling plume depend on the spatial distribution of the radionuclides in the air, the energy of the radiation, and the extent of shielding. The AXAIR89Q code takes no credit for shielding in calculating doses. The code calculates gamma doses using a nonuniform Gaussian model, which has more realistic modeling than doses from the conventional uniform semi-infinite plume model.

In addition to using the worst sector, 99.5 percentile meteorology, conservative breathing rates, and taking no credit for shielding, the AXAIR89Q code also takes no credit for the probable plume rise from stack releases. Therefore, the offsite maximum individual doses calculated by AXAIR89Q provide conservative bounding estimates of radiological consequences to exposed individuals and populations from postulated accidental atmospheric releases.

AXAIR89Q has been validated for compliance to accepted standards for such software. Attachment A, Accident Analysis, discusses AXAIR89Q and its predecessor, AXAIR. When used in conjunction with models for predicting health effects, the results from AXAIR89Q can be compared with other site-specific codes such as RSAC-5, because both codes provide relative radionuclide concentrations based on the guidance provided in NRC Regulatory Guide 1.145.

This section summarizes the potential for radiological accidents and their consequences for the cases under each alternative. Attachment A describes the methodology and assumptions used in the assessment; describes radiological accident scenarios in more detail; provides source terms and references used to estimate the doses and impacts for each alternative and case; and includes scaling factors that the DOE decisionmaker can apply to the source term or dose for each facility associated with a case.

DOE assessed the potential impacts from a selected spectrum of radiological release accidents, ranging from low (1×10^{-6} event per year) to high (more than 1 event per year) frequencies of occurrence, along with the associated impacts (doses and potential latent fatal cancers) that could result. The accidents used as references are attributed to individual facilities based on their functions and processes (see Attachment A, Table A-3), not to specific cases or alternatives. This enables a comparison of alternatives depending on which facilities support a specific case or alternative. Figure 5-1 is a flowchart for the preparation of accident analysis information. No new analyses occurred because existing documentation adequately supports a quantitative or qualitative estimation of potential impacts, as required by the National Environmental Policy Act of 1969. The assessment of postulated radiological accidents associated with spent nuclear fuel at the SRS indicates that the highest point estimate of risk to the public within 80 kilometers (50 miles) of the Site would be 1.4×10^{-3} latent fatal cancer per year. The estimated dose to the same population from all causes, including natural background sources, would be about 19,000 person-rem per year (DOE 1990), which could cause about nine latent fatal cancers per year in the same population. For perspective, natural background radiation sources would result in approximately 6,000 times the risk associated with the largest consequence accident postulated in this EIS for the various spent nuclear fuel management alternatives.

DOE did not quantitatively analyze the potential health effects for SRS workers less than 100 meters (328 feet) from radiological accidents. Computer codes used to calculate radiological doses can experience potentially large errors as a source disperses throughout a building. However, DOE did carry out a qualitative evaluation of the potential radiological effects to SRS workers in the immediate vicinity of an accident related to spent fuel management. DOE estimates that the consequences of an accident for the most part would result in higher than normal radiation doses. However, no fatalities would occur except in the event of an inadvertent criticality in FB-Line, where up to four fatalities may result. This evaluation is discussed in more detail in Section A.2-6.2 of Attachment A.

5.15.2.1 Alternative 1 - No Action. This alternative identifies the minimum actions deemed necessary for continued safe and secure management of spent nuclear fuel at the SRS. As explained in Chapter 3, this is not a *status quo* condition. Spent nuclear fuel would be maintained close to defueling or current storage locations with minimal facility upgrade or equipment replacement. Only local transport would occur. SRS activities required to safely store spent nuclear fuel would continue. This alternative would require SRS to place corroded and pitted fuel elements in cans to minimize spread of material into the pool. DOE estimated potential radiological accident impacts that could occur under this alternative using existing DOE-approved safety analyses for the interim wet storage of

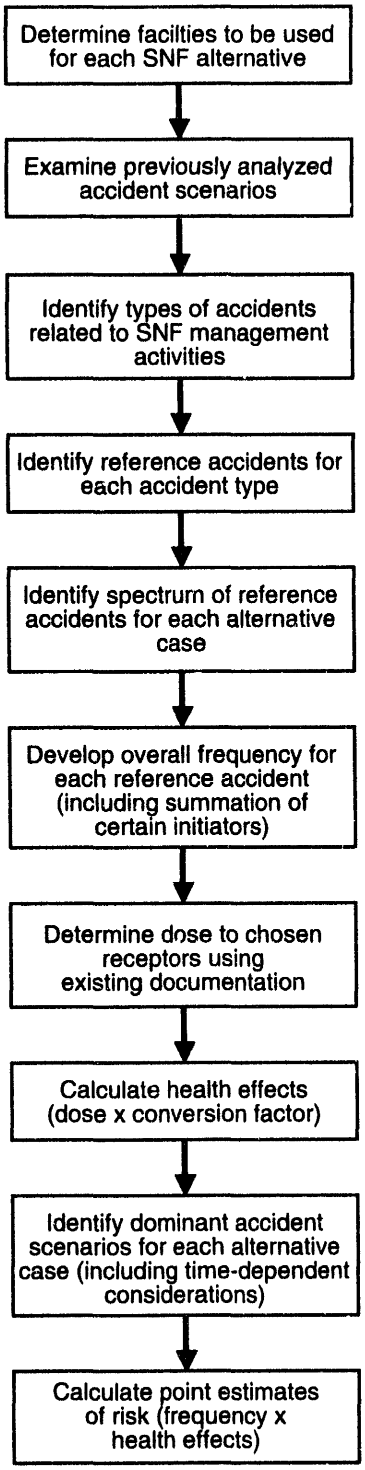


Figure 5-1. Accident analysis process.

spent nuclear fuel at SRS facilities. As indicated in Attachment A, Table A-3, the facilities required under this alternative would consist of existing facilities, including necessary upgrades to support safe interim wet storage. In addition, Attachment A, Table A-4, provides a reference accident spectrum associated with these facilities for this alternative. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this alternative, as well as their estimated frequencies. Table 5-20 lists the accident scenario with the highest point estimates of risk to the general public. Table 5-21 compares the potential radiological accidents and health effects of the interim wet storage (Option 1) of spent nuclear fuel for the No-Action alternative.

Table 5-20. Highest point estimates of risk among receptor groups (Option 1).

	Receptor Groups	
	Maximally Exposed Offsite Individual	Population to 80 kilometers
Overall Point Estimate of Risk[a]	1.6×10^{-7} (Fuel Assembly Breach)	1.4×10^{-3} (Fuel Assembly Breach)

a. Units of latent fatal cancers per year.

5.15.2.2 Alternative 2 - Decentralization. Accident assessments considered for this alternative include those considered for the no-action alternative for wet storage (Option 2b) plus assessments for the dry storage (Option 2a) of spent nuclear fuel and for the processing of spent fuel (Option 2c). Option 2c (processing) assumes the use of existing facilities to dissolve, separate, and further stabilize spent nuclear fuel. For cases that include some treatment (e.g., canning) of spent nuclear fuel, such treatment is referred to as "stabilization," not processing. The amount of fuel of various types to be considered would include those quantities from the production reactors, existing research fuel, foreign research reactor fuel, and fuel transported for safety or research activities.

5.15.2.2.1 Option 2a - Dry Storage — DOE estimated potential radiological accident impacts that could occur in this case using existing DOE-approved safety analysis reports submitted to DOE by Westinghouse Savannah River Company for vault storage of special nuclear material from existing facilities. DOE has not incorporated the technology to support interim dry storage of spent nuclear fuel at the SRS. To provide a basis for evaluating the potential impacts from this alternative case, this assessment used data from existing safety analyses for special nuclear material storage facilities and extrapolated these data to apply to spent nuclear fuel. DOE also considered radiological accidents associated with wet storage, at least in the near term, because the spent nuclear fuel is

Table 5-21. Radioactive release accidents and health effects for spent nuclear fuel alternatives.[a,b]

Alternative (by case)	Accident Scenario	Frequency (per year)	Potential Fatal Cancers: Maximally exposed offsite individual[d]	Potential Fatal Cancers: Population to 80 kilometers[d]	Potential Fatal Cancers: Worker[e]	Potential Fatal Cancers: Colocated Worker[e]	Point Estimate of Risk[c]: Maximally exposed offsite individual	Point Estimate of Risk[c]: Population to 80 kilometers[f]	Point Estimate of Risk[c]: Worker	Point Estimate of Risk[c]: Colocated Worker
1. No Action										
Option 1 Wet Storage	A1 Fuel Assembly Breach	1.6×10^{-1}	1.0×10^{-6}	8.5×10^{-3}	(a)	4.8×10^{-6}	1.6×10^{-7}	1.4×10^{-3}	(a)	7.7×10^{-7}
	A4 Material Release (Adjacent Facility)	2.4×10^{-3}	3.0×10^{-6}	2.5×10^{-2}	(a)	2.0×10^{-5}	7.2×10^{-9}	6.0×10^{-5}	(a)	4.8×10^{-8}
	A5 Criticality in Water	3.1×10^{-3}	1.5×10^{-6}	4.4×10^{-3}	(a)	5.6×10^{-5}	4.7×10^{-9}	1.4×10^{-5}	(a)	1.7×10^{-7}
	A7 Spill/Liquid Discharge (external)	7.9×10^{-3}	8.5×10^{-6}	1.4×10^{-5}	(a)	(b)	6.7×10^{-8}	1.1×10^{-7}	(a)	(b)
	A8 Spill/Liquid Discharge (internal)	1.1×10^{-1}	1.2×10^{-13}	1.0×10^{-9}	(a)	8.0×10^{-15}	1.3×10^{-14}	1.1×10^{-10}	(a)	8.8×10^{-16}
2. Decentralization										
Option 2a Dry Storage	A1 Fuel Assembly Breach	1.6×10^{-1}	1.0×10^{-6}	8.5×10^{-3}	(a)	4.8×10^{-6}	1.6×10^{-7}	1.4×10^{-3}	(a)	7.7×10^{-7}
	A3 Material Release (Dry Vault)	1.4×10^{-3}	1.1×10^{-9}	3.5×10^{-6}	(a)	(b)	1.5×10^{-12}	4.9×10^{-9}	(a)	(b)
	A4 Material Release (Adjacent Facility)	2.4×10^{-3}	3.0×10^{-6}	2.5×10^{-2}	(a)	2.0×10^{-5}	7.2×10^{-9}	6.0×10^{-5}	(a)	4.8×10^{-8}
	A5 Criticality in Water	3.1×10^{-3}	1.5×10^{-6}	4.4×10^{-3}	(a)	5.6×10^{-5}	4.7×10^{-9}	1.4×10^{-5}	(a)	1.7×10^{-7}
	A7 Spill/Liquid Discharge (external)	7.9×10^{-3}	8.5×10^{-6}	1.4×10^{-5}	(a)	(b)	6.7×10^{-8}	1.1×10^{-7}	(a)	(b)
	A8 Spill/Liquid Discharge (internal)	1.1×10^{-1}	1.2×10^{-13}	1.0×10^{-9}	(a)	8.0×10^{-15}	1.3×10^{-14}	1.1×10^{-10}	(a)	8.8×10^{-16}
Option 2b Wet Storage	A1 Fuel Assembly Breach	1.6×10^{-1}	1.0×10^{-6}	8.5×10^{-3}	(a)	4.8×10^{-6}	1.6×10^{-7}	1.4×10^{-3}	(a)	7.7×10^{-7}
	A4 Material Release (Adjacent Facility)	2.4×10^{-3}	3.0×10^{-6}	2.5×10^{-2}	(a)	2.0×10^{-5}	7.2×10^{-9}	6.0×10^{-5}	(a)	4.8×10^{-8}
	A5 Criticality in Water	3.1×10^{-3}	1.5×10^{-6}	4.4×10^{-3}	(a)	5.6×10^{-5}	4.7×10^{-9}	1.4×10^{-5}	(a)	1.7×10^{-7}
	A7 Spill/Liquid Discharge (external)	7.9×10^{-3}	8.5×10^{-6}	1.4×10^{-5}	(a)	(b)	6.7×10^{-8}	1.1×10^{-7}	(a)	(b)
	A8 Spill/Liquid Discharge (internal)	1.1×10^{-1}	8.2×10^{-13}	1.0×10^{-9}	(a)	8.0×10^{-15}	1.3×10^{-14}	1.1×10^{-10}	(a)	8.8×10^{-16}
Option 2c Processing	A1 Fuel Assembly Breach	1.6×10^{-1}	1.0×10^{-6}	8.5×10^{-3}	(a)	4.8×10^{-6}	1.6×10^{-7}	1.4×10^{-3}	(a)	7.7×10^{-7}
	A2 Material Release (Processing)	5.0×10^{0}	8.5×10^{-11}	6.5×10^{-7}	1.1×10^{-2}	(b)	4.3×10^{-10}	3.3×10^{-6}	5.4×10^{-2}	(b)

Table 5-21. (continued).

Alternative (by case)	Accident Scenario	Frequency (per year)	Potential Fatal Cancers: Maximally exposed offsite individual[d]	Potential Fatal Cancers: Population to 80 kilometers[d]	Potential Fatal Cancers: Worker[e]	Potential Fatal Cancers: Colocated Worker[e]	Point Estimate of Risk[c]: Maximally exposed offsite individual	Point Estimate of Risk[c]: Population to 80 kilometers[f]	Point Estimate of Risk[c]: Worker	Point Estimate of Risk[c]: Colocated Worker
Option 2c (continued)	A3 Material Release (Dry Vault)	1.4×10^{-3}	1.1×10^{-9}	3.5×10^{-6}	(a)	(b)	1.5×10^{-12}	4.9×10^{-9}	(a)	(b)
	A4 Material Release (Adjacent Facility)	2.4×10^{-3}	3.0×10^{-6}	2.5×10^{-2}	(a)	2.0×10^{-5}	7.2×10^{-9}	6.0×10^{-5}	(a)	4.8×10^{-8}
	A5 Criticality in Water	3.1×10^{-3}	1.5×10^{-6}	4.4×10^{-3}	(a)	5.6×10^{-5}	4.7×10^{-9}	1.4×10^{-5}	(a)	1.7×10^{-7}
	A6 Criticality in Processing	1.4×10^{-4}	1.3×10^{-6}	1.5×10^{-3}	9.6×10^{-1}	(b)	1.8×10^{-10}	2.1×10^{-7}	1.3×10^{-4}	(b)
	A7 Spill/Liquid Discharge (external)	7.9×10^{-3}	8.5×10^{-6}	1.4×10^{-5}	(a)	(b)	6.7×10^{-8}	1.1×10^{-7}	(a)	(b)
	A8 Spill/Liquid Discharge (internal)	1.1×10^{-3}	1.2×10^{-13}	1.0×10^{-9}	(a)	8.0×10^{-15}	1.2×10^{-14}	1.1×10^{-10}	(a)	8.8×10^{-16}
3. 1992/1993 Planning Basis										
Option 3a Dry Storage	Same as Option 2a for Decentralization									
Option 3b Wet Storage	Same as Option 2b for Decentralization									
Option 3c Processing	Same as Option 2c for Decentralization									
4. Regionalization - A										
Option 4a Dry Storage	Same as Option 2a for Decentralization									
Option 4b Wet Storage	Same as Option 2b for Decentralization									
Option 4c Processing	Same as Option 2c for Decentralization									

Table 5-21. (continued).

Alternative (by case)	Accident Scenario	Frequency (per year)	Potential Fatal Cancers: Maximally exposed offsite individual[d]	Potential Fatal Cancers: Population to 80 kilometers[d]	Potential Fatal Cancers: Worker[e]	Potential Fatal Cancers: Colocated Worker[e]	Point Estimate of Risk[c]: Maximally exposed offsite individual	Point Estimate of Risk[c]: Population to 80 kilometers[f]	Point Estimate of Risk[c]: Worker	Point Estimate of Risk[c]: Colocated Worker
4. Regionalization - B										
Option 4d Dry Storage				Same as Option 2a for Decentralization						
Option 4e Wet Storage				Same as Option 2b for Decentralization						
Option 4f Processing				Same as Option 2c for Decentralization						
Option 4g Shipping Out				Same as Option 1 for No Action						
5. Centralization										
Option 5a Dry Storage				Same as Option 2a for Decentralization						
Option 5b Wet Storage				Same as Option 2b for Decentralization						
Option 5c Processing				Same as Option 2c for Decentralization						
Option 5d Shipping Out				Same as Option 1 No Action						

a. The safety analysis reports from which information was extracted for these accidents were written before the issuance of DOE Order 5480.23; previous Orders did not require the inclusion of workers.
b. The safety analysis reports from which information was extracted for these accidents were written before the issuance of DOE Order 5480.23; previous Orders did not require the inclusion of colocated workers.
c. Units for point estimates of risk are given in potential latent fatal cancers per year.
d. ICRP 60 risk fact.r for the general public (5.0×10^{-4} fatal cancer per year) was used to determine potential latent fatal cancers.
e. ICRP 60 risk factor for workers (4.0×10^{-4} fatal cancer per year) was used to determine potential latent fatal cancers.

currently in wet storage. Similarly, this assessment includes fuel handling accidents throughout the transition phase (i.e., until fuel is in interim dry storage). As indicated in Attachment A, Table A-4, the facilities required under this alternative would consist of existing and new facilities necessary to support the safe handling, stabilization, and dry storage of spent nuclear fuel. In addition, Table A-4 identifies a potential accident spectrum associated with these facilities for this case. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this alternative case, as well as the estimated frequency of occurrence for each accident. Table 5-21 lists the potential radiological accidents and health effects associated with dry storage of spent nuclear fuel for the Decentralization alternative. For the transition period of wet to dry storage, Table 5-22 lists the accident scenario with the highest overall point estimate of risk to the general public. Table 5-22 lists the accident scenario with the highest point estimate of risk (after transition) to the general public when the fuel had been moved from wet storage (after approximately 15 years) and placed in interim dry storage. This indicates a substantial reduction in risk (more than six orders of magnitude) when fuel handling events are no longer potential accident initiators.

Table 5-22. Highest point estimates of risk among receptor groups (Option 2a).

	Receptor Groups	
	Maximally Exposed Offsite Individual	Population to 80 kilometers
Overall Point Estimate of Risk[a]	1.6×10^{-7} (Fuel Assembly Breach)	1.4×10^{-3} (Fuel Assembly Breach)
Transitioned to Dry Storage Point Estimate of Risk[a]	1.5×10^{-12} (Dry Vault Material Release)	4.9×10^{-9} (Dry Vault Material Release)

a. Units of latent fatal cancers per year.

5.15.2.2.2 Option 2b - Wet Storage — DOE estimated potential radiological accident impacts that could occur under this case using existing DOE-approved safety analysis reports and amendments submitted to DOE by Westinghouse Savannah River Company for existing wet storage facilities. As indicated in Attachment A, Table A-4, the facilities (modules as defined in the WSRC 1994b and Figure 3-1) would consist of existing facilities and specific upgrades necessary to support safe interim wet storage. In addition, Table A-4 identifies the reference accident spectrum associated with these facilities for this option. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this alternative option, as well as the estimated frequency of occurrence for each accident. Table 5-21 lists the radiological accidents and consequences of the wet storage (Option 2b) of spent nuclear fuel for the Decentralization alternative. Table 5-23 lists the accident scenario with the highest point estimate of risk to the general public. For wet pool storage options, there are no transition phases.

Table 5-23. Highest point estimates of risk among receptor groups (Option 2b).

	Receptor Groups	
	Maximally Exposed Offsite Individual	Population to 80 kilometers
Overall Point Estimate of Risk[a]	1.6×10^{-7} (Fuel Assembly Breach)	1.4×10^{-3} (Fuel Assembly Breach)

a. Units of latent fatal cancers per year.

5.15.2.2.3 Option 2c - Processing and Storage — Processing for the SRS is defined as the operation of the separations facilities in F- or H-Areas. The H-Area facilities were designed to recover uranium and plutonium from spent production reactor fuel, and the F-Area facilities were designed to recover plutonium.

DOE estimated potential radiological accident impacts that could occur under this option using existing DOE-approved safety analysis reports submitted to DOE by Westinghouse Savannah River Company for processes and for vault storage of special nuclear material from existing facilities. DOE also considered radiological accidents associated with wet storage, because the spent nuclear fuel is currently in wet storage. Similarly, it included fuel handling accidents throughout the processing phase (i.e., until special nuclear material is in interim dry storage). As indicated in Attachment A, Table A-4, the facilities required under this option would consist of existing and new facilities necessary to support safe handling and processing of spent nuclear fuel into special nuclear material for dry storage. In addition, Table A-4 identifies the reference accident spectrum associated with these facilities for this case. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this alternative case, as well as the estimated frequency of occurrence for each accident. Table 5-21 lists the radiological release accidents and health effects for the processing of spent nuclear fuel to special nuclear material for the Decentralization alternative. Table 5-24 lists the accident scenario with the highest overall point estimate of risk to the general public from the transition period of wet spent fuel storage into processing for special nuclear material. When the fuel had been processed from wet storage to special nuclear material and placed in its interim dry storage, Table 5-24 lists the accident scenario with the highest point estimate of risk after

Table 5-24. Highest point estimates of risk among receptor groups (Option 2c).

	Receptor Groups	
	Maximally Exposed Offsite Individual	Population to 80 kilometers[b]
Overall Point Estimate of Risk[a]	1.6×10^{-7} (Fuel Assembly Breach)	1.4×10^{-3} (Fuel Assembly Breach)
Transitioned to Dry Storage Point Estimate of Risk[a]	1.5×10^{-12} (Dry Vault Material Release)	4.9×10^{-9} (Dry Vault Material Release)

a Units of latent fatal cancers per year.

transition to the general public. This indicates a substantial reduction in risk (more than six orders of magnitude) when fuel handling events and processing events are no longer potential accident initiators.

For this option, DOE assumes it could not process some fuel clad in stainless steel or zirconium into special nuclear material and, therefore, would dry-store it as fuel. The technology for dry storage of nonaluminum-clad fuel has been demonstrated and is assumed to pose no greater risk than monitored dry storage of special nuclear material.

5.15.2.3 Alternative 3 - 1992/1993 Planning Basis. Because this alternative would be consistent with the *status quo* at the SRS, existing documents contain sufficient information to examine its accident analysis impacts. The SRS would continue to receive the spent nuclear fuel designated for the Site, and DOE would complete facilities already planned to accommodate the existing inventory and the spent nuclear fuel receipts. This alternative would require the same facilities already used to support the cases discussed in the Section 5.15.2.2. The major difference would be the amount of fuel ultimately stored because this alternative assumes the continued receipt of fuel beyond that shipped to the SRS under the Decentralization alternative.

5.15.2.3.1 Option 3a - Dry Storage — DOE estimated potential radiological accident impacts that could occur under this case using existing DOE-approved safety analysis reports for vault storage from existing facilities and the study discussed for Option 2a. DOE also considered radiological accidents associated with wet storage, at least in the near term, because the spent nuclear fuel is currently in wet storage. Similarly, it included fuel handling accidents throughout the transition phase (i.e., until the fuel is in interim dry storage). As indicated in Attachment A, Table A-4, the facilities required under this option would consist of existing and new facilities necessary to support the safe handling and stabilization of spent nuclear fuel for dry storage. In addition, Table A-4 identifies the reference accident spectrum associated with these facilities for this case. Attachment A, Table A-2, lists the authorization basis references for the source terms considered in analyzing potential accidents under this option, as well as the estimated frequency of occurrence for each accident. Table 5-21 lists the radiological release accidents and health effects for the dry storage of spent nuclear fuel for the 1992/1993 Planning Basis alternative. For the entire period, the accident scenarios with the highest point estimates of risk to the general public would be the same as those for Option 2a, as listed in Table 5-22.

5.15.2.3.2 Option 3b - Wet Storage — DOE estimated potential radiological accident impacts that could occur under this case using existing DOE-approved safety analysis reports and from amendments submitted to DOE by Westinghouse Savannah River Company for wet storage for existing facilities. As indicated in Attachment A, Table A-4, the facilities required under this option would consist of existing facilities and upgrades necessary to support safe interim wet storage. In addition, Table A-4 identifies the reference accident spectrum associated with these facilities for this option. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this option, as well as the estimated frequency of occurrence for each accident. Table 5-21 lists the radiological release accidents and health effects of the wet storage (Option 2b) of spent nuclear fuel for the 1992/1993 Planning Basis alternative. The accident scenario with the highest point estimate of risk to the general public would be the same as that for Option 2b, as listed in Table 5-23.

5.15.2.3.3 Option 3c - Processing and Storage. Table 5-21 lists the radioactive release accidents and health effects for the processing of spent nuclear fuel for this option. After processing is complete, the accident scenario with the highest point estimate of risk would be associated with the storage of special nuclear materials, as discussed for Option 2c and listed in Table 5-24.

5.15.2.4 Alternative 4 - Regionalization. This alternative comprises Regionalization A and Regionalization B subalternatives. Under the Regionalization A subalternative (Options 4a, 4b, and 4c), the SRS would receive all aluminum-clad fuel from the other sites considered in this EIS and would transfer its existing inventory of stainless-steel- and Zircaloy-clad fuel to other DOE sites, as appropriate. These proposed activities would reflect current and past activities, so sufficient information and analyses are available to enable the scaling or other extrapolation of radiological accident impacts. The total amount of spent nuclear fuel to be managed under Regionalization A would be slightly less than that for Alternatives 2 and 3; the decisionmaker could use this amount to adjust the estimated point estimate of risk by the use of an appropriate adjustment (scaling) factor, as discussed in Attachment A, Section A.2.9.

Under the Regionalization B subalternative (Options 4d, 4e, 4f, and 4g), the SRS would receive all existing and new spent nuclear fuel east of the Mississippi River. The decisionmaker could use the change in spent nuclear fuel inventories to adjust the estimated point estimate of risk by the use of an appropriate adjustment (scaling) factor, as discussed in Attachment A, Section A.2.9. For the purposes of this evaluation, Option 4g (Section 5.15.2.4.7) assumes that DOE would ship all fuel off the Site to the Oak Ridge Reservation.

5.15.2.4.1 Option 4a - Dry Storage — This case is similar to Option 2a, with the exception of the quantity and type of fuel to be stored. As with Option 2a, this assessment evaluated existing analyses; the point estimates of risk are the same as those for Option 2a.

5.15.2.4.2 Option 4b - Wet Storage — This case is similar to Option 2b, with the exception of a slightly smaller quantity of fuel to be stored. As with Option 2b, this assessment evaluated existing analyses, and the point estimates of risk are the same as those for Option 2b.

5.15.2.4.3 Option 4c - Processing and Storage — For this option, the accident analysis evaluation is similar to Option 2c. DOE assumes that it could process spent nuclear fuel associated with regionalization at SRS with existing facilities, because they are designed to process aluminum-clad fuel. However, the small amount of aluminum-clad fuel received after major processing options are completed would be placed in wet storage.

5.15.2.4.4 Option 4d - Dry Storage — The accident analysis evaluation for this option is similar to that for Option 2a, with the exception of the increased inventories and types of fuel to be stored.

5.15.2.4.5 Option 4e - Wet Storage — The accident analysis evaluation for this option is similar to that for Option 2b, with the exception of the increased inventories and types of fuel to be stored.

5.15.2.4.6 Option 4f - Processing and Storage — For this option, the accident analysis evaluation is similar to Option 2c. DOE assumes that it could process all the current SRS aluminum-clad spent nuclear fuel with existing facilities. However, all receipts of spent nuclear fuel will be placed in dry storage as discussed for Option 4d.

5.15.2.4.7 Option 4g - Shipping Off Site — This option assumes that DOE would characterize the fuel and ship it all off the Site. Thus, the potential radiological accidents considered are the same as those for Alternative 1.

5.15.2.5 Alternative 5 - Centralization. This alternative for the SRS would involve fuel types and new facilities beyond those considered for any other alternative. For instance, under this alternative, the SRS would receive spent nuclear fuel from the U.S. Navy. One of the new facilities that would be necessary to support this type of spent nuclear fuel is the Expended Core Facility (ECF). Volume 1, Appendix D, includes a detailed accident analyses for this proposed facility using SRS-specific parameters.

This alternative would bound the maximum number of spent nuclear fuel-related accident scenarios that DOE could expect at the SRS, due to the number of new facilities at the Site that would have to accommodate the diversity and the increased amount of the fuel to be managed. The decisionmaker could use this maximum amount of spent nuclear fuel to adjust the estimated risk by the use of an appropriate scaling factor, as discussed in Attachment A, Section A.2.9. For the purposes of this evaluation, Option 5d (Section 5.15.2.5.4) assumes that DOE would ship all fuel off the Site to another DOE facility.

5.15.2.5.1 Option 5a - Dry Storage — The major difference in dry storage facilities between this alternative and the others would be the addition of a facility for Naval spent nuclear fuels and the large quantity of spent fuel shipped to the SRS from the Hanford Site. DOE estimated potential radiological accident impacts that could occur under this option using DOE-approved safety analysis reports submitted to DOE by Westinghouse Savannah River Company for vault storage in existing facilities at the SRS and the study discussed for Option 2a. In addition, DOE considered radiological accidents associated with wet storage, at least in the near term, because the SRS spent nuclear fuel is currently in wet storage. Similarly, it included fuel handling accidents throughout the transition phase (i.e., until fuel is in interim dry storage). As indicated in Attachment A, Table A-4, the facilities required under this option would consist of existing and new facilities necessary to support the safe handling and stabilization of spent nuclear fuel for dry storage. In addition,

Table A-4 identifies the reference accident spectrum associated with these facilities for this case. Attachment A, Table A-2, lists the references for the source terms considered in analyzing potential accidents under this option, as well as the estimated frequency of occurrence for each accident. Table 5-21 compares the radiological release accidents and health effects for the dry storage of spent nuclear fuel for the Centralization alternative. From the transition period of wet to dry storage, the accident scenario with the highest point estimate of risk to the general public would be the same as that for Option 2a, as listed in Table 5-22. When the fuel had been moved from wet storage (after approximately 25 years) and placed in interim dry storage, the accident scenario with the highest point estimate of risk to the population would be the same as the Option 2a dry storage phase.

5.15.2.5.2 Option 5b - Wet Storage — The accident analysis evaluation for this option is similar to that for Option 2b, with the exception of the amount and type of fuel to be stored.

5.15.2.5.3 Option 5c - Processing and Storage — For this option, the accident analysis evaluation is similar to Option 2c. DOE assumes that it could process the current SRS aluminum-clad spent nuclear fuel with existing facilities. However, the SRS would place all receipts of fuel in dry storage, as discussed for Option 5a.

5.15.2.5.4 Option 5d - Shipping Off Site — This option assumes that DOE would perform the characterization of the fuel at the SRS, and then would ship all fuel off the Site. Thus, the potential radiological accidents considered are the same as those for the No-Action alternative.

5.15.3 Chemical Hazard Evaluation

For toxic chemicals, several government agencies recommend the quantification of health effects as threshold values of concentrations in air or water that cause short-term effects. The long-term health consequences of human exposure to toxic chemicals are not as well understood as those for radiation. Thus, the potential health effects from toxic chemicals are more subjective than those from radioactive materials.

This section provides a quantitative discussion for an analyzed chemical accident at the Receiving Basin for Offsite Fuel facility and qualitative discussions addressing chemical hazards for each of the other existing SRS facilities involved in the receipt, processing, transport, or storage of spent nuclear fuel.

5.15.3.1 Receiving Basin for Offsite Fuel. The maximum reasonably foreseeable chemical hazard accident for the Receiving Basin for Offsite Fuel would involve the release of nitrogen dioxide vapor following the complete reaction of a drum of target cleaning solution (13.4 percent nitric acid) with sodium nitrite (WSRC 1993b). The initiator for this accident is a leak from a storage tank into the target cleaning solution and involves multiple failures or maloperations with an accident probability comparable to that of a natural phenomena accident. Table 5-25 shows the concentration of nitrogen dioxide vapor that an individual at the SRS boundary and a maximally exposed colocated worker could receive.

Table 5-25. Results of analyzed chemical accident.

Receptor Group	Frequency (per year)	NO_2 Concentration (mg/m^3)
Maximally Exposed Offsite Individual	1.0×10^{-3}	0.083
Colocated Worker	1.0×10^{-3}	0.64

To determine the potential health effects from this bounding chemical accident scenario, this assessment was to compare the resulting airborne concentrations of nitrogen dioxide at various receptor distances against Emergency Response Planning Guideline (ERPG) values, where available. Because there were no ERPG values available for nitrogen dioxide, the assessment substituted other chemical toxicity values as follows:

- For Emergency Response Planning Guideline 1, the assessment substituted threshold limit values/time-weighted average (TLV/TWA) values (ACGIH 1987). The time-weighted average is the average concentration for a normal 8-hour workday and a 40-hour workweek from which nearly all workers could receive repeated exposure, day-after-day, without adverse effect.

- For Emergency Response Planning Guideline 2, the assessment substituted level of concern (LOC) values [equal to 0.1 of the immediately dangerous to life or health (IDLH) value; - see below]. The level of concern value is the concentration of a hazardous substance in the air above which there could be serious irreversible health effects or death as a result of a single exposure for a relatively short period of time (EPA 1987).

- For Emergency Response Planning Guideline 3, the assessment substituted immediately dangerous to life or health values. This value is the maximum concentration from which a person could escape within 30 minutes without a respirator and without experiencing any impairment of escape or irreversible side effects (NIOSH 1990).

These values as they apply to nitrogen dioxide are as follows:

- Time-weighted average value = 5.6 milligrams per cubic meter
- Level of concern value = 9.4 milligrams per cubic meter
- Immediately dangerous to life or health value = 94.0 milligrams per cubic meter

5.15.3.2 Reactor Basins. There are no postulated chemical accidents for the reactor basins that would cause an impact to an individual at the SRS boundary or a colocated worker.

5.15.3.3 H-Area. There are no postulated chemical accidents for the H-Area Canyon that would cause an impact to an individual at the SRS boundary or a colocated worker. DOE has performed an accident analysis for the H-Area Canyon facility workers that indicates the existence of potential injuries due to chemical contamination or exposure to hazardous vapors at or above the level of concern exposure limit (Du Pont 1983a). The analysis does not project exposure to hazardous vapors at or above the immediate danger to life and health level to occur.

The probability that a worker could be accidentally exposed to any of the hazardous liquids identified in Attachment A, Table A-14, is bounded by a frequency of 2.8×10^{0} per year (Du Pont 1983a). The most likely injury is an acid burn to the skin.

The probability for exposure to hazardous vapors at or above the level of concern exposure limit is 8.5×10^{-1} per year (Du Pont 1983a). The potential for chemical uptakes and for illness would depend on the safety measures taken before the exposure, the duration of the exposure, and the mitigating actions taken after the exposure.

5.15.3.4 F-Area. There are no postulated chemical accidents for the F-Area Canyon that would cause an impact to an individual at the SRS boundary or a colocated worker. DOE has performed an accident analysis for the F-Area Canyon facility workers that indicates the existence of potential injuries due to chemical contamination or exposure to hazardous vapors at or above the level of concern exposure limit (Du Pont 1983b). The analysis does not project exposure to hazardous vapors at or above the immediate danger to life and health level to occur.

The probability that a worker could be accidentally exposed to any one of the hazardous liquids identified in Attachment A, Table A-15, is bounded by a frequency of 1.2 x 10^0 per year (Du Pont 1983b). The most likely injury is an acid burn to the skin.

The probability for exposure to hazardous vapors at or above the level of concern exposure limit is 3.2 x 10^{-1} per year (Du Pont 1983b). The potential for chemical uptakes and for illness would depend on the safety measures taken before the exposure, the duration of the exposure, and the mitigating actions taken after the exposure.

5.15.4 Secondary Impacts

The primary focus of the accident analysis is to determine the magnitude of the consequences of postulated accident scenarios on public and worker health and safety. However, DOE recognizes that chemical and radiological accidents can also adversely affect the surrounding environment (i.e., secondary impacts). Accordingly, DOE has qualitatively evaluated each of the eight radiological accident scenarios considered in this analysis for potential secondary impacts. The following paragraphs discuss the results of the evaluation, and Table 5-26 summarizes expected secondary impacts for each accident scenario.

5.15.4.1 Biotic Resources. With the exception of a direct discharge of disassembly basin water to an onsite stream, DOE does not expect radiological contamination resulting from any of the analyzed accidents to reach any onsite or offsite surface water. DOE previously evaluated the case of a direct discharge of disassembly basin water (DOE 1990) and believes that impacts on biotic resources would be minor. Therefore, the impacts on aquatic biota from any of the accident scenarios would be minor. Small areas of minor surface contamination likely would be outside the industrialized area of a postulated accident. Terrestrial biota in or near the contaminated area would be exposed to small quantities of radioactive materials and ionizing radiation until the affected area could be decontaminated. DOE believes that the impacts on biotic resources from this exposure would be minor.

5.15.4.2 Water Resources. DOE expects no adverse impacts on water quality from any of the postulated accident scenarios. Accident A7 - External Spill/Liquid Discharge would be expected

Table 5-26. Qualitative summary of expected secondary impacts.

Accident Scenario	Accident Description	Environmental or social factor: Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
A1	Fuel assembly breach	No adverse effects on biota expected.	No adverse effects expected to surface or groundwater resources.	Limited economic impacts are expected. Any required cleanup could be handled with existing workforce.	No effect.	Local contamination expected around site of the accident. Minor contamination outside the immediate facility area unlikely to require cleanup of more than 10 acres.	No impacts expected.	No change expected. No irreversible impacts.	No impact to Native American or public lands expected.
A2	Material release (processing)	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A3	Material release (dry vault)	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A4	Material release (adjacent facility)	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A5	Criticality in water	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A6	Criticality during processing	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A7	External spill/liquid discharge	Same as A1.	Surface-water table contamination expected in area of the release. No adverse effects expected to surface-water or drinking water aquifers.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.	Same as A1.
A8	Internal spill/liquid discharge	Same as A1.	No adverse impact to water resources. The spill is expected to be contained entirely within the building structure.	Same as A1.	Same as A1.	Limited contamination is expected outside the effected building.	Same as A1.	Same as A1.	Same as A1.

to have the most significant impact. With the exception of the reactor disassembly basins, the location and configuration of existing or potential facilities would prevent a direct release of radionuclide-contaminated water to surface water. However, contamination of the surface aquifer in the area of the release would be likely. The processes governing the slow plume movement and attenuation of contaminants described in Section 5.8 would prevent the contamination from reaching surface- or groundwater resources. Similarly, radionuclide contamination of onsite or offsite drinking water sources would be unlikely. DOE evaluated the effects of a direct discharge of disassembly basin water on water resources (DOE 1990) and believes that impacts on water resources would be minimal.

5.15.4.3 Economic Impacts. DOE expects limited economic impacts as a result of any of the postulated accidents. Any cleanup required would be localized, and the existing workforce and equipment could perform it. Contamination should be contained within a small area inside the SRS boundaries for all eight postulated accident scenarios. The existing workforce could accomplish any required cleanup.

5.15.4.4 National Defense. None of the postulated accidents would affect the DOE national defense mission. Spent nuclear fuel management activities do not involve the production of materials needed for national defense.

5.15.4.5 Environmental Contamination. DOE expects that none of the postulated accident scenarios would result in large areas of contamination. Local contamination is likely around the site of an accident, but in all scenarios should be contained within the SRS boundaries. Minor contamination outside the immediate area of the accident is unlikely to require cleanup of more than a small area inside the Site boundary. Impacts in all cases should be minimal.

5.15.4.6 Endangered Species. There are no Federally listed threatened or endangered species habitats in the immediate vicinity of existing or potential spent nuclear fuel storage or processing facilities (see Section 4.9.4). None of the postulated accident scenarios would likely result in large areas of surface contamination outside the immediate facilities, and DOE does not expect adverse impacts to surface water. Therefore, none of the postulated accident scenarios is likely to impact threatened or endangered species.

5.15.4.7 Land Use. No accident scenario should result in large areas of contamination, nor would the impacts be irreversible. DOE expects no change in land use.

5.15.4.8 Treaty Rights. The environmental impacts of each of the accident scenarios should be contained within the SRS boundaries. Because there are no Native American or public lands within the site boundaries, treaty rights would not be affected.

5.15.5 Adjusted Point Estimate of Risk Summary

The accident scenarios described in Section 5.15.2 differ only slightly between the various alternatives. These scenarios did not account for variations in spent nuclear fuel shipments (including onsite operational transfers) and spent fuel storage inventories across the alternatives. To provide a realistic comparison across alternatives, DOE developed adjustment factors to adjust frequencies or consequences, depending on the specific circumstance of each alternative. Attachment A, Section A.2.9, provides the methodology and justifications used to develop appropriate adjustment factors. This section provides the adjusted point estimates of risk for each accident scenario by receptor group to demonstrate a relative comparison of each alternative on a case-by-case basis. Tables 5-27, 5-28, and 5-29 summarize the adjusted point estimates of risk for each alternative for the maximally exposed individual, the general population to 80 kilometers, and the colocated worker.

5.16 Cumulative Impacts

The Savannah River Site (SRS) contains major U.S. Department of Energy (DOE) and non-DOE facilities, unrelated to spent nuclear fuel, that would continue to operate throughout the life of the spent nuclear fuel management program. The activities associated with these existing facilities produce environmental consequences that this document has included in the baseline environmental conditions (Chapter 4) against which it assesses the consequences of the spent nuclear fuel alternatives. Impacts of both the construction and operation of SRS spent nuclear fuel facilities would be cumulative with the impacts of existing and planned facilities unrelated to spent nuclear fuel.

This cumulative impact assessment considered the incremental and synergistic effects of the operation of the Defense Waste Processing Facility, which is nearing completion, and the Consolidated Incineration Facility, which is under construction, when appropriate and when data existed. For example, the Air Quality analysis factored in emissions from these two facilities when considering potential impacts of operations of spent nuclear fuel facilities. The small volumes of liquid effluent (treated sanitary wastes) currently entering the environment from the Defense Waste Processing Facility, on the other hand, were considered part of the Water Quality baseline. The only major stand

Table 5-27. Adjusted point estimates of risk for the maximally exposed offsite individual (radiological accidents).

		No Action	Decentralization			92/93 Planning Basis			Regionalization - A			Centralization			
Accident Description	Attribute[d]	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c	Option 5a	Option 5b	Option 5c	Option 5d
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}
	Adjusted Annual Frequency	1.6×10^{-1}	3.3×10^{-1}	3.5×10^{-1}	1.6×10^{-1}	4.0×10^{-1}	4.0×10^{-1}	2.3×10^{-1}	4.4×10^{-1}	4.4×10^{-1}	2.8×10^{-1}	8.4×10^{-1}	8.4×10^{-1}	6.8×10^{-1}	1.7×10^{-1}
	Adjusted Point Estimate of Risk[b]	1.6×10^{-7}	3.3×10^{-7}	3.5×10^{-7}	1.6×10^{-7}	4.0×10^{-7}	4.0×10^{-7}	2.3×10^{-7}	4.4×10^{-7}	4.4×10^{-7}	2.8×10^{-7}	8.4×10^{-7}	8.4×10^{-7}	6.8×10^{-7}	1.7×10^{-7}
A2 - Processing release	Adjusted Health Effects[a]	(c)	(c)	(c)	8.5×10^{-11}	(c)	(c)	8.5×10^{-11}	(c)	(c)	8.5×10^{-11}	(c)	(c)	8.5×10^{-11}	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	5.2×10^{0}	(c)	(c)	5.3×10^{0}	(c)	(c)	5.2×10^{0}	(c)	(c)	6.9×10^{1}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	4.5×10^{-10}	(c)	(c)	4.5×10^{-10}	(c)	(c)	4.4×10^{-10}	(c)	(c)	5.9×10^{-9}	(c)
A3 - Dry vault release	Adjusted Health Effects[a]	(c)	1.1×10^{-9}	(c)	1.1×10^{-9}	1.2×10^{-9}	(c)	1.2×10^{-9}	1.1×10^{-9}	(c)	1.1×10^{-9}	1.5×10^{-8}	(c)	1.5×10^{-8}	(c)
	Adjusted Annual Frequency	(c)	1.4×10^{-3}	(c)	1.4×10^{-3}	1.4×10^{-3}	(c)	1.4×10^{-3}	1.4×10^{-3}	(c)	1.4×10^{-3}	1.4×10^{-3}	(c)	1.4×10^{-3}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	1.6×10^{-12}	(c)	1.6×10^{-12}	1.6×10^{-12}	(c)	1.6×10^{-12}	1.5×10^{-12}	(c)	1.5×10^{-12}	2.1×10^{-11}	(c)	2.1×10^{-11}	(c)
A4 - Adjacent facility release	Adjusted Health Effects[a]	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}
	Adjusted Annual Frequency	2.4×10^{-3}	5.0×10^{-3}	5.3×10^{-3}	2.5×10^{-3}	5.9×10^{-3}	5.9×10^{-3}	3.4×10^{-3}	6.6×10^{-3}	6.6×10^{-3}	4.2×10^{-3}	1.3×10^{-2}	1.3×10^{-2}	1.0×10^{-2}	2.5×10^{-3}
	Adjusted Point Estimate of Risk[b]	7.2×10^{-9}	1.5×10^{-8}	1.6×10^{-8}	7.4×10^{-8}	1.8×10^{-8}	1.8×10^{-8}	1.0×10^{-8}	2.0×10^{-8}	2.0×10^{-8}	1.3×10^{-8}	3.8×10^{-8}	3.8×10^{-8}	3.0×10^{-8}	7.4×10^{-9}

Table 5-27. (continued).

		No Action	Decentralization			92/93 Planning Basis			Regionalization - A			Centralization			
Accident Description	Attribute[d]	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c	Option 5a	Option 5b	Option 5c	Option 5d
A5 - Criticality in water	Adjusted Health Effect[a]	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$	$1.5x10^{-6}$
	Adjusted Annual Frequency	$3.1x10^{-3}$	$6.4x10^{-3}$	$6.8x10^{-3}$	$3.2x10^{-3}$	$7.7x10^{-3}$	$7.7x10^{-3}$	$4.4x10^{-3}$	$8.6x10^{-3}$	$8.6x10^{-3}$	$5.5x10^{-3}$	$1.6x10^{-2}$	$1.6x10^{-2}$	$1.3x10^{-2}$	$3.3x10^{-3}$
	Adjusted Point Estimate of Risk[b]	$4.7x10^{-9}$	$9.7x10^{-9}$	$1.0x10^{-8}$	$4.8x10^{-9}$	$1.2x10^{-8}$	$1.2x10^{-8}$	$6.7x10^{-9}$	$1.3x10^{-8}$	$1.3x10^{-8}$	$8.3x10^{-9}$	$2.5x10^{-8}$	$2.5x10^{-8}$	$2.0x10^{-8}$	$5.0x10^{-9}$
A6 - Criticality during processing	Adjusted Health Effects[a]	(c)	(c)	(c)	$1.3x10^{-6}$	(c)	(c)	$1.3x10^{-6}$	(c)	(c)	$1.3x10^{-6}$	(c)	(c)	$1.3x10^{-6}$	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	$1.5x10^{-4}$	(c)	(c)	$1.5x10^{-4}$	(c)	(c)	$1.4x10^{-4}$	(c)	(c)	$1.9x10^{-3}$	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	$1.9x10^{-10}$	(c)	(c)	$1.9x10^{-10}$	(c)	(c)	$1.9x10^{-10}$	(c)	(c)	$2.5x10^{-9}$	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[a]	$8.5x10^{-6}$	$8.8x10^{-6}$	$8.8x10^{-6}$	$8.8x10^{-6}$	$8.9x10^{-6}$	$8.9x10^{-6}$	$8.9x10^{-6}$	$8.8x10^{-6}$	$8.8x10^{-6}$	$8.8x10^{-6}$	$1.2x10^{-4}$	$1.2x10^{-4}$	$1.2x10^{-4}$	$1.2x10^{-4}$
	Adjusted Annual Frequency	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$	$7.9x10^{-3}$
	Adjusted Point Estimate of Risk[b]	$6.7x10^{-8}$	$7.0x10^{-8}$	$7.0x10^{-8}$	$7.0x10^{-8}$	$7.0x10^{-8}$	$7.0x10^{-8}$	$7.0x10^{-8}$	$7.0x10^{-8}$	$7.0x10^{-8}$	$7.0x10^{-8}$	$9.5x10^{-7}$	$9.5x10^{-7}$	$9.5x10^{-7}$	$9.5x10^{-7}$
A8 - Internal spill/liquid discharge	Adjusted Health Effects[a]	$1.2x10^{-13}$	$1.2x10^{-13}$	$1.2x10^{-13}$	$1.2x10^{-13}$	$1.3x10^{-13}$	$1.3x10^{-13}$	$1.3x10^{-13}$	$1.2x10^{-13}$	$1.2x10^{-13}$	$1.2x10^{-13}$	$1.6x10^{-12}$	$1.6x10^{-12}$	$1.6x10^{-12}$	$1.6x10^{-12}$
	Adjusted Annual Frequency	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$	$1.1x10^{-1}$
	Adjusted Point Estimate of Risk[b]	$1.3x10^{-4}$	$1.3x10^{-4}$	$1.3x10^{-4}$	$1.3x10^{-4}$	$1.4x10^{-14}$	$1.4x10^{-14}$	$1.4x10^{-14}$	$1.3x10^{-14}$	$1.3x10^{-14}$	$1.3x10^{-14}$	$1.8x10^{-13}$	$1.8x10^{-13}$	$1.8x10^{-13}$	$1.3x10^{-14}$

a. Units for adjusted health effects are given in terms of potential fatal cancers.
b. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
c. The accident scenario is not included in the spectrum of potential accidents for this case.
d. Adjustment factors were calculated using March 1994 data and information. In-process revisions to these data and information should not result in changes to these factors by more than 10 percent.

Table 5-27. (continued).

Accident Description	Attribute[a]	Regionalization - B Option 4d	Option 4e	Option 4f	Option 4g
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}
	Adjusted Annual Frequency	4.1×10^{-1}	4.1×10^{-1}	2.5×10^{-1}	1.7×10^{-1}
	Adjusted Point Estimate of Risk[b]	4.1×10^{-7}	4.1×10^{-7}	2.5×10^{-7}	1.7×10^{-7}
A2 - Processing release	Adjusted Health Effects[a]	(c)	(c)	8.5×10^{-11}	(c)
	Adjusted Annual Frequency	(c)	(c)	6.6×10^{0}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	5.6×10^{-10}	(c)
A3 - Dry vault release	Adjusted Health Effects[a]	1.4×10^{-9}	(c)	1.4×10^{-9}	(c)
	Adjusted Annual Frequency	1.4×10^{-3}	(c)	1.4×10^{-3}	(c)
	Adjusted Point Estimate of Risk[b]	2.0×10^{-12}	(c)	2.0×10^{-12}	(c)
A4 - Adjacent facility release	Adjusted Health Effects[a]	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}
	Adjusted Annual Frequency	6.2×10^{-3}	6.2×10^{-3}	3.7×10^{-3}	2.5×10^{-3}
	Adjusted Point Estimate of Risk[b]	1.9×10^{-8}	1.9×10^{-8}	1.1×10^{-8}	7.5×10^{-9}

Table 5-27. (continued).

Accident Description	Attribute[d]	Regionalization - B Option 4d	Option 4e	Option 4f	Option 4g
A5 - Criticality in water	Adjusted Health Effect[a]	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}
	Adjusted Annual Frequency	8.0×10^{-3}	8.0×10^{-3}	4.8×10^{-3}	3.3×10^{-3}
	Adjusted Point Estimate of Risk[b]	1.2×10^{-8}	1.2×10^{-8}	7.2×10^{-9}	4.9×10^{-9}
A6 - Criticality during processing	Adjusted Health Effects[a]	(c)	(c)	1.3×10^{-6}	(c)
	Adjusted Annual Frequency	(c)	(c)	1.8×10^{-4}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	2.4×10^{-10}	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[a]	1.1×10^{-5}	1.1×10^{-5}	1.1×10^{-5}	1.1×10^{-5}
	Adjusted Annual Frequency	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}
	Adjusted Point Estimate of Risk[b]	8.7×10^{-8}	8.7×10^{-8}	8.7×10^{-8}	8.7×10^{-8}
A8 - Internal spill/liquid discharge	Adjusted Health Effects[a]	1.6×10^{-13}	1.6×10^{-13}	1.6×10^{-13}	1.6×10^{-13}
	Adjusted Annual Frequency	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}
	Adjusted Point Estimate of Risk[b]	1.7×10^{-14}	1.7×10^{-14}	1.7×10^{-14}	1.7×10^{-14}

a. Units for adjusted health effects are given in terms of potential fatal cancers.
b. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
c. The accident scenario is not included in the spectrum of potential accidents for this case.
d. Adjustment factors were calculated using March 1994 data and information. In-process revisions to these data and information should not result in changes to these factors by more than 10 percent.

Table 5-28. Adjusted point estimates of risk for the colocated worker (radiological accidents).

		No Action	Decentralization			92/93 Planning Basis			Regionalization - A			Centralization			
Accident Description	Attribute	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c	Option 5a	Option 5b	Option 5c	Option 5d
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}
	Adjusted Annual Frequency	1.6×10^{-1}	3.3×10^{-1}	3.5×10^{-1}	1.6×10^{-1}	4.0×10^{-1}	4.0×10^{-1}	2.3×10^{-1}	4.4×10^{-1}	4.4×10^{-1}	2.8×10^{-1}	8.4×10^{-1}	8.4×10^{-1}	6.8×10^{-1}	1.7×10^{-1}
	Adjusted Point Estimate of Risk[b]	7.7×10^{-7}	1.6×10^{-6}	1.7×10^{-6}	7.7×10^{-7}	1.9×10^{-6}	1.9×10^{-6}	1.1×10^{-6}	2.1×10^{-6}	2.1×10^{-6}	1.3×10^{-6}	4.0×10^{-6}	4.0×10^{-6}	3.3×10^{-6}	8.2×10^{-7}
A2 - Processing release	Adjusted Health Effects[a]	(c)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)
A3 - Dry vault release	Adjusted Health Effects[a]	(c)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(c)
	Adjusted Annual Frequency	(c)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(d)	(c)	(d)	(c)
A4 - Adjacent facility release	Adjusted Health Effects[a]	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}
	Adjusted Annual Frequency	2.4×10^{-3}	5.0×10^{-3}	5.3×10^{-3}	2.5×10^{-3}	5.9×10^{-3}	5.9×10^{-3}	3.4×10^{-3}	6.6×10^{-3}	6.6×10^{-3}	4.2×10^{-3}	1.3×10^{-2}	1.3×10^{-2}	1.0×10^{-2}	2.5×10^{-3}
	Adjusted Point Estimate of Risk[b]	4.8×10^{-8}	1.0×10^{-7}	1.1×10^{-7}	4.9×10^{-8}	1.2×10^{-7}	1.2×10^{-7}	6.8×10^{-8}	1.3×10^{-7}	1.3×10^{-7}	8.5×10^{-8}	2.5×10^{-7}	2.5×10^{-7}	2.0×10^{-7}	5.0×10^{-8}
A5 - Criticality in water	Adjusted Health Effects[a]	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}
	Adjusted Annual Frequency	3.1×10^{-3}	6.4×10^{-3}	6.8×10^{-3}	3.2×10^{-3}	7.7×10^{-3}	7.7×10^{-3}	4.4×10^{-3}	8.6×10^{-3}	8.6×10^{-3}	5.5×10^{-3}	1.6×10^{-2}	1.6×10^{-2}	1.3×10^{-2}	3.3×10^{-3}
	Adjusted Point Estimate of Risk[b]	1.7×10^{-7}	3.6×10^{-7}	3.8×10^{-7}	1.8×10^{-7}	4.3×10^{-7}	4.3×10^{-7}	2.5×10^{-7}	4.8×10^{-7}	4.8×10^{-7}	3.1×10^{-7}	9.0×10^{-7}	9.0×10^{-7}	7.3×10^{-7}	1.8×10^{-7}

Table 5-28. (continued).

		No Action	Decentralization			92/93 Planning Basis			Regionalization - A			Centralization			
Accident Description	Attribute	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c	Option 5a	Option 5b	Option 5c	Option 5d
A6 - Criticality during processing	Adjusted Health Effects[a]	(c)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)	(c)	(d)	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[a]	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)
	Adjusted Annual Frequency	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)
	Adjusted Point Estimate of Risk[b]	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)
A8 - Internal spill/liquid discharge	Adjusted Health Effects[a]	8.0×10^{-15}	8.3×10^{-15}	8.3×10^{-15}	8.3×10^{-15}	8.4×10^{-15}	8.4×10^{-15}	8.4×10^{-15}	8.2×10^{-15}	8.2×10^{-15}	8.2×10^{-15}	1.1×10^{-13}	1.1×10^{-13}	1.1×10^{-13}	1.1×10^{-13}
	Adjusted Annual Frequency	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}
	Adjusted Point Estimate of Risk[b]	8.8×10^{-16}	9.2×10^{-16}	9.2×10^{-16}	9.2×10^{-16}	9.2×10^{-16}	9.2×10^{-16}	9.2×10^{-16}	9.1×10^{-16}	9.1×10^{-16}	9.1×10^{-16}	1.2×10^{-14}	1.2×10^{-14}	1.2×10^{-14}	1.2×10^{-14}

a. Units for adjusted health effects are given in terms of potential fatal cancers.
b. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
c. The accident scenario is not included in the spectrum of potential accidents for this case.
d. The safety analyses from which information was extracted for these accidents were written before issuance of DOE Order 5480.23; previous Orders did not require the inclusion of colocated workers.

Table 5-28. (continued).

Accident Description	Attribute	Regionalization - B Option 4d	Option 4e	Option 4f	Option 4g
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}	4.8×10^{-6}
	Adjusted Annual Frequency	4.1×10^{-1}	4.1×10^{-1}	2.5×10^{-1}	1.7×10^{-1}
	Adjusted Point Estimate of Risk[b]	2.0×10^{-6}	2.0×10^{-6}	1.2×10^{-6}	8.1×10^{-7}
A2 - Processing release	Adjusted Health Effects[a]	(c)	(c)	(d)	(c)
	Adjusted Annual Frequency	(c)	(c)	(d)	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(d)	(c)
A3 - Dry vault release	Adjusted Health Effects[a]	(c)	(c)	(d)	(c)
	Adjusted Annual Frequency	(c)	(c)	(d)	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(d)	(c)
A4 - Adjacent facility release	Adjusted Health Effects[a]	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}
	Adjusted Annual Frequency	6.2×10^{-3}	6.2×10^{-3}	3.7×10^{-3}	2.5×10^{-3}
	Adjusted Point Estimate of Risk[b]	1.2×10^{-7}	1.2×10^{-7}	7.4×10^{-7}	5.0×10^{-8}

Table 5-28. (continued).

Accident Description	Attribute	Regionalization - B Option 4d	Option 4e	Option 4f	Option 4g
A5 - Criticality in water	Adjusted Health Effects[a]	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}	5.6×10^{-5}
	Adjusted Annual Frequency	8.0×10^{-3}	8.0×10^{-3}	4.8×10^{-3}	3.3×10^{-3}
	Adjusted Point Estimate of Risk[b]	4.5×10^{-7}	4.5×10^{-7}	2.7×10^{-7}	1.8×10^{-7}
A6 - Criticality during processing	Adjusted Health Effects[a]	(c)	(c)	(d)	(c)
	Adjusted Annual Frequency	(c)	(c)	(d)	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(d)	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[a]	(c)	(c)	(d)	(c)
	Adjusted Annual Frequency	(c)	(c)	(d)	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(d)	(c)
A8 - Internal spill/liquid discharge	Adjusted Health Effects[a]	1.0×10^{-14}	1.0×10^{-14}	1.0×10^{-14}	1.0×10^{-14}
	Adjusted Annual Frequency	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}
	Adjusted Point Estimate of Risk[b]	1.2×10^{-15}	1.2×10^{-15}	1.2×10^{-15}	1.2×10^{-15}

a. Units for adjusted health effects are given in terms of potential fatal cancers.
b. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
c. The accident scenario is not included in the spectrum of potential accidents for this case.
d. The safety analyses from which information was extracted for these accidents were written before issuance of DOE Order 5480.23; previous Orders did not require the inclusion of colocated workers.

Table 5-29. Adjusted point estimates of risk for the general population - 80 kilometers (radiological accidents).

		No Action	Decentralization			92/93 Planning Basis			Regionalization - A			Centralization			
Accident Description	Attribute	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c	Option 5a	Option 5b	Option 5c	Option 5d
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}
	Adjusted Annual Frequency	1.6×10^{-1}	3.3×10^{-1}	3.5×10^{-1}	1.6×10^{-1}	4.0×10^{-1}	4.0×10^{-1}	2.3×10^{-1}	4.4×10^{-1}	4.4×10^{-1}	2.8×10^{-1}	8.4×10^{-1}	8.4×10^{-1}	6.8×10^{-1}	1.7×10^{-1}
	Adjusted Point Estimate of Risk[b]	1.4×10^{-3}	2.8×10^{-3}	3.0×10^{-3}	1.4×10^{-3}	3.4×10^{-3}	3.4×10^{-3}	2.0×10^{-3}	3.7×10^{-3}	3.7×10^{-3}	2.4×10^{-3}	7.2×10^{-3}	7.2×10^{-3}	5.8×10^{-3}	1.4×10^{-3}
A2 - Processing release	Adjusted Health Effects[a]	(c)	(c)	(c)	6.5×10^{-7}	(c)	(c)	6.5×10^{-7}	(c)	(c)	6.5×10^{-7}	(c)	(c)	6.5×10^{-7}	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	5.2×10^{0}	(c)	(c)	5.3×10^{0}	(c)	(c)	5.2×10^{0}	(c)	(c)	6.9×10^{1}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	3.4×10^{-6}	(c)	(c)	3.5×10^{-6}	(c)	(c)	3.4×10^{-6}	(c)	(c)	4.5×10^{-5}	(c)
A3 - Dry vault release	Adjusted Health Effects[a]	(c)	3.6×10^{-6}	(c)	3.6×10^{-6}	3.7×10^{-6}	(c)	3.7×10^{-6}	3.6×10^{-6}	(c)	3.6×10^{-6}	4.8×10^{-5}	(c)	4.8×10^{-5}	(c)
	Adjusted Annual Frequency	(c)	1.4×10^{-3}	(c)	1.4×10^{-3}	1.4×10^{-3}	(c)	1.4×10^{-3}	1.4×10^{-3}	(c)	1.4×10^{-3}	1.4×10^{-3}	(c)	1.4×10^{-3}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	5.0×10^{-9}	(c)	5.0×10^{-9}	5.0×10^{-9}	(c)	5.1×10^{-9}	5.0×10^{-9}	(c)	5.0×10^{-9}	6.7×10^{-8}	(c)	6.7×10^{-8}	(c)
A4 - Adjacent facility release	Adjusted Health Effects[a]	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}	2.5×10^{-2}
	Adjusted Annual Frequency	2.4×10^{-3}	5.0×10^{-3}	5.3×10^{-3}	2.5×10^{-3}	5.9×10^{-3}	5.9×10^{-3}	3.4×10^{-3}	6.6×10^{-3}	6.6×10^{-3}	4.2×10^{-3}	1.3×10^{-2}	1.3×10^{-2}	1.0×10^{-2}	2.5×10^{-3}
	Adjusted Point Estimate of Risk[b]	6.0×10^{-5}	1.2×10^{-4}	1.3×10^{-4}	6.2×10^{-5}	1.5×10^{-4}	1.5×10^{-4}	8.5×10^{-5}	1.7×10^{-4}	1.7×10^{-4}	1.1×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	2.5×10^{-4}	6.2×10^{-5}

Table 5-29. (continued).

		No Action	Decentralization			92/93 Planning Basis			Regionalization - A			Centralization			
Accident Description	Attribute	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c	Option 4a	Option 4b	Option 4c	Option 5a	Option 5b	Option 5c	Option 5d
A5 - Criticality in water	Adjusted Health Effects[a]	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-2}	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-3}
	Adjusted Annual Frequency	3.1×10^{-3}	6.4×10^{-3}	6.8×10^{-3}	3.2×10^{-3}	7.7×10^{-3}	7.7×10^{-3}	4.4×10^{-3}	8.6×10^{-3}	8.6×10^{-3}	5.5×10^{-3}	1.6×10^{-2}	1.6×10^{-2}	1.3×10^{-2}	3.3×10^{-3}
	Adjusted Point Estimate of Risk[b]	1.4×10^{-5}	2.8×10^{-4}	3.0×10^{-4}	1.4×10^{-4}	3.4×10^{-4}	3.4×10^{-4}	1.9×10^{-4}	3.8×10^{-4}	3.8×10^{-4}	2.4×10^{-4}	7.0×10^{-4}	7.0×10^{-5}	5.7×10^{-4}	1.5×10^{-5}
A6 - Criticality during processing	Adjusted Health Effects[a]	(c)	(c)	(c)	1.5×10^{-3}	(c)	(c)	1.5×10^{-3}	(c)	(c)	1.5×10^{-3}	(c)	(c)	1.5×10^{-3}	(c)
	Adjusted Annual Frequency	(c)	(c)	(c)	1.5×10^{-4}	(c)	(c)	1.5×10^{-4}	(c)	(c)	1.4×10^{-4}	(c)	(c)	1.9×10^{-3}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	(c)	2.2×10^{-7}	(c)	(c)	2.2×10^{-7}	(c)	(c)	2.2×10^{-7}	(c)	(c)	2.9×10^{-6}	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[a]	1.4×10^{-5}	1.5×10^{-5}	1.5×10^{-5}	1.5×10^{-5}	1.5×10^{-5}	1.5×10^{-5}	1.5×10^{-5}	1.4×10^{-5}	1.4×10^{-5}	1.4×10^{-5}	1.9×10^{-4}	1.9×10^{-4}	1.9×10^{-4}	1.9×10^{-4}
	Adjusted Annual Frequency	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}
	Adjusted Point Estimate of Risk[b]	1.1×10^{-7}	1.1×10^{-7}	1.1×10^{-7}	1.1×10^{-7}	1.1×10^{-7}	1.1×10^{-7}	1.1×10^{-7}	1.1×10^{-7}	1.1×10^{-7}	1.1×10^{-7}	1.5×10^{-7}	1.5×10^{-7}	1.5×10^{-7}	1.5×10^{-7}
A8 - Internal spill/liquid discharge	Adjusted Health Effects[a]	1.0×10^{-9}	1.0×10^{-9}	1.0×10^{-9}	1.0×10^{-9}	1.1×10^{-9}	1.1×10^{-9}	1.1×10^{-9}	1.0×10^{-9}	1.0×10^{-9}	1.0×10^{-9}	1.4×10^{-8}	1.4×10^{-8}	1.4×10^{-8}	1.4×10^{-8}
	Adjusted Annual Frequency	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}
	Adjusted Point Estimate of Risk[b]	1.1×10^{-10}	1.1×10^{-10}	1.1×10^{-10}	1.1×10^{-10}	1.2×10^{-10}	1.2×10^{-10}	1.2×10^{-10}	1.1×10^{-10}	1.1×10^{-10}	1.1×10^{-10}	1.5×10^{-9}	1.5×10^{-9}	1.5×10^{-9}	1.5×10^{-9}

a. Units for adjusted health effects are given in terms of potential fatal cancers.
b. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
c. The accident scenario is not included in the spectrum of potential accidents for this case.

Table 5-29. (continued).

Accident Description	Attribute	Regionalization - B Option 4d	Option 4e	Option 4f	Option 4g
A1 - Fuel Assembly Breach	Adjusted Health Effects[a]	$8.5x10^{-3}$	$8.5x10^{-3}$	$8.5x10^{-3}$	$8.5x10^{-3}$
	Adjusted Annual Frequency	$4.1x10^{-1}$	$4.1x10^{-1}$	$2.5x10^{-1}$	$1.7x10^{-1}$
	Adjusted Point Estimate of Risk[b]	$3.5x10^{-3}$	$3.5x10^{-3}$	$2.1x10^{-3}$	$1.4x10^{-3}$
A2 - Processing Release	Adjusted Health Effects[a]	(c)	(c)	$6.5x10^{-7}$	(c)
	Adjusted Annual Frequency	(c)	(c)	$6.6x10^{0}$	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	$4.3x10^{-6}$	(c)
A3 - Dry vault Release	Adjusted Health Effects[a]	$4.6x10^{-6}$	(c)	$4.6x10^{-6}$	(c)
	Adjusted Annual Frequency	$1.4x10^{-3}$	(c)	$1.4x10^{-3}$	(c)
	Adjusted Point Estimate of Risk[b]	$6.4x10^{-4}$	(c)	$6.4x10^{-4}$	(c)
A4 - Adjacent Facility Release	Adjusted Health Effects[a]	$2.5x10^{-2}$	$2.5x10^{-2}$	$2.5x10^{-2}$	$2.5x10^{-2}$
	Adjusted Annual Frequency	$6.2x10^{-3}$	$6.2x10^{-3}$	$3.7x10^{-3}$	$2.5x10^{-3}$
	Adjusted Point Estimate of Risk[b]	$1.6x10^{-4}$	$1.6x10^{-4}$	$9.2x10^{-5}$	$6.3x10^{-5}$

Table 5-29. (continued).

Accident Description	Attribute	Regionalization - B Option 4d	Option 4e	Option 4f	Option 4g
A5 - Criticality in water	Adjusted Health Effects[a]	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-3}	4.4×10^{-3}
	Adjusted Annual Frequency	8.0×10^{-3}	8.0×10^{-3}	4.8×10^{-3}	3.3×10^{-3}
	Adjusted Point Estimate of Risk[b]	3.5×10^{-5}	3.5×10^{-5}	2.1×10^{-5}	1.4×10^{-5}
A6 - Criticality during processing	Adjusted Health Effects[a]	(c)	(c)	1.5×10^{-3}	(c)
	Adjusted Annual Frequency	(c)	(c)	1.8×10^{-4}	(c)
	Adjusted Point Estimate of Risk[b]	(c)	(c)	2.8×10^{-7}	(c)
A7 - External spill/liquid discharge	Adjusted Health Effects[a]	1.8×10^{-5}	1.8×10^{-5}	1.8×10^{-5}	1.8×10^{-5}
	Adjusted Annual Frequency	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}	7.9×10^{-3}
	Adjusted Point Estimate of Risk[b]	1.4×10^{-7}	1.4×10^{-7}	1.4×10^{-7}	1.4×10^{-7}
A8 - Internal spill/liquid discharge	Adjusted Health Effects[a]	1.3×10^{-9}	1.3×10^{-9}	1.3×10^{-9}	1.3×10^{-9}
	Adjusted Annual Frequency	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}
	Adjusted Point Estimate of Risk[b]	1.4×10^{-10}	1.4×10^{-10}	1.4×10^{-10}	1.4×10^{-10}

a. Units for adjusted health effects are given in terms of potential fatal cancers.
b. Units for adjusted point estimates of risk are given in terms of potential fatal cancers per year.
c. The accident scenario is not included in the spectrum of potential accidents for this case.

alone facilities scheduled to be built in the near future on the SRS are the Savannah River Ecology Laboratory Conference Center and the new Centralized Sanitary Wastewater Treatment Facility. A number of other planned facilities have not been factored into the cumulative impacts analysis because final funding approval has not been received or because decisions on these facilities involve major unresolved DOE policy issues. For example, this cumulative impact assessment does not consider long-term reconfiguration issues. Table 5-30 presents a summary of cumulative impacts associated with the various spent fuel management alternatives.

5.16.1 Land Use

The land committed to spent nuclear fuel management activities at the SRS would lie, for the most part, within existing onsite industrial compounds or undeveloped onsite areas devoted to the continued mission of the Site. Under two of the alternatives - Regionalization by Location (at SRS) and Centralization (at SRS) - a new Expended Core Facility could be required to examine and characterize spent nuclear fuels from naval installations east of the Mississippi. Two locations have been proposed for the Expended Core Facility, one in the approximate center of the SRS and the other at the old Allied General Nuclear Services facility (or "Barnwell Nuclear Fuel Plant") that is located off Road G (and near SRS Barricade 4) just east of and adjacent to the Site.

Previously-undeveloped land committed to new spent nuclear fuel facilities (excluding the Expended Core Facility) would be limited to a maximum of approximately 100 acres (0.4 square kilometer). Depending on the location chosen, an additional 30 acres (0.1 square kilometer) could be required for a new Expended Core Facility. Thus, a maximum of 130 acres (0.5 square kilometer) could be converted from woodlands or old fields to industrial facilities and supporting infrastructure under the bounding options, Option 5a (Centralization - Dry Storage) and Option 5c (Centralization - Processing). Any site used for the support of spent nuclear fuel activities would be under government control. With the exception of the Barnwell Nuclear Fuel facility, which the Navy would purchase from Allied General Nuclear Services for an offsite Expended Core Facility, DOE would not require any additional land from the public domain for SRS spent nuclear fuel management facilities.

Ground was broken for the new Savannah River Ecology Laboratory Conference Center in May 1994. The new facility will occupy a 70-acre area, but only 5 to 10 acres will be cleared and graded for the new conference center, parking areas, and an access road. The remaining 60-65 acres will be managed as a nature study area and preserve. Thus, the Savannah River Ecology Laboratory

Table 5-30. Cumulative impacts associated with construction and operation of spent fuel alternatives at Savannah River Site.

ALTERNATIVE 1 - NO ACTION

	Option 1 Wet Storage
Land Use	No new land committed to new use.
Socioeconomics	A maximum of 50 new jobs created annually during construction; no new jobs created during operation.
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be $9.0x10^{-5}$ rem.
Materials and Waste Management	High-Level: Current generation levels Transuranic: Current generation levels Low-Level: Current generation levels Mixed: Current generation levels Hazardous: Current generation levels Sanitary: Current generation levels

ALTERNATIVE 2 - DECENTRALIZATION

	Option 2a Dry Storage	Option 2b Wet Storage	Option 2c Processing
Land Use	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.
Socioeconomics	Construction jobs: 600 peak Operation: No new jobs	Construction jobs: 600 peak Operation: No new jobs	Construction jobs: 550 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be $9.0x10^{-5}$ rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be $9.0x10^{-5}$ rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be $4.4x10^{-4}$ rem.
Materials and Waste Management	High-Level: No change Transuranic: 6% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: No change Transuranic: 6% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: Small increase[c] Transuranic: 18% increase Low-Level: 425% increase Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]

Table 5-30. (continued).

ALTERNATIVE 3 - 1992/1993 PLANNING BASIS			
	Option 3a Dry Storage	Option 3b Wet Storage	Option 3c Processing
Land Use	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.
Socioeconomics	Construction jobs: 600 peak Operation: No new jobs	Construction jobs: 650 peak Operation: No new jobs	Construction jobs: 550 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 4.5×10^{-4} rem.
Materials and Waste Management	High-Level: No change Transuranic: 6% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: No change Transuranic: 6% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: Small increase[c] Transuranic: 18% increase Low-Level: 425% increase Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]

ALTERNATIVE 4 - REGIONALIZATION			
	Option 4a Dry Storage	Option 4b Wet Storage	Option 4c Processing
Land Use	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.	Small amount of land (<10 acres) committed to new use.
Socioeconomics	Construction jobs: 650 peak Operation: No new jobs	Construction jobs: 650 peak Operation: No new jobs	Construction jobs: 550 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 4.7×10^{-4} rem.

Table 5-30. (continued).

	Option 4a Dry Storage		Option 4b Wet Storage		Option 4c Processing	
Materials and Waste Management	High-Level:	No change	High-Level:	No change	High-Level:	Small increase[c]
	Transuranic:	6% increase	Transuranic:	6% increase	Transuranic:	18% increase
	Low-Level:	No change	Low-Level:	No change	Low-Level:	425% increase
	Mixed:	No change[a]	Mixed:	No change[a]	Mixed:	No change[a]
	Hazardous:	No change[a]	Hazardous:	No change[a]	Hazardous:	No change[a]
	Sanitary:	No change[b]	Sanitary:	No change[b]	Sanitary:	No change[b]

	Option 4d Dry Storage		Option 4e Wet Storage		Option 4f Processing	
Land Use	Approximately 40 acres committed to new use.		Approximately 35 acres committed to new use.		Approximately 35 acres committed to new use.	
Socioeconomics	Construction jobs:	910 peak	Construction jobs:	910 peak	Construction jobs:	860 peak
	Operation:	No new jobs	Operation:	No new jobs	Operation:	No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.		Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.		Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.		Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.		Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 4.7×10^{-4} rem.	
Materials and Waste Management	High-Level:	No change	High-Level:	No change	High-Level:	Small increase[c]
	Transuranic:	6% increase	Transuranic:	6% increase	Transuranic:	18% increase
	Low-Level:	No change	Low-Level:	No change	Low-Level:	425% increase
	Mixed:	No change[a]	Mixed:	No change[a]	Mixed:	No change[a]
	Hazardous:	No change[a]	Hazardous:	No change[a]	Hazardous:	No change[a]
	Sanitary:	No change[b]	Sanitary:	No change[b]	Sanitary:	No change[b]

	Option 4g Ship Out
Land Use	Less than one acre of land committed to new use.
Socioeconomics	Construction jobs: 200 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be (less than) $<9.0 \times 10^{-5}$ rem.

Table 5-30. (continued).

Materials and Waste Management	High-Level:	No change
	Transuranic:	Reduced volume of waste produced
	Low-Level:	No change
	Mixed:	No change[a]
	Hazardous:	No change[a]
	Sanitary:	No change[b]

ALTERNATIVE 5 - CENTRALIZATION

	Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing
Land Use	100-130 acres of land committed to new use.	70-80 acres of land committed to new use.	100-130 acres of land committed to new use.
Socioeconomics	Construction: 2,550 peak Operation: No new jobs	Construction: 2,700 peak Operation: No new jobs	Construction: 2,550 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 4.7×10^{-4} rem.
Materials and Waste Management	High-Level: No change Transuranic: 6% increase Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: No change Transuranic: 18% increase Low-Level: 25% increase Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]	High-Level: Small increase[c] Transuranic: 18% increase Low-Level: 425% increase Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]

	Option 5d Ship Out
Land Use	Less than one acre of land committed to new use.
Socioeconomics	Construction: 200 peak Operation: No new jobs
Air Resources	Site emissions would not exceed any air quality standard. Table 5-31 lists cumulative Site nonradioactive releases at the SRS boundary.
Occupational and Public Health and Safety	Radioactive airborne releases, expressed as cumulative dose to a maximally exposed individual at the Site boundary, would be 9.0×10^{-5} rem.
Materials and Waste Management	High-Level: No change Transuranic: Reduced volume of wastes produced Low-Level: No change Mixed: No change[a] Hazardous: No change[a] Sanitary: No change[b]

a. Not expected to change; no analysis conducted.
b. Not expected to change; based on projected employment levels at SRS.
c. Small increase (an average of 2 cubic meters per year) from zero baseline.

Conference Center will require conversion of 5 to 10 acres of planted pines or pine/mixed hardwood (depending on the exact location of the building) to light-industrial/public use.

Construction on the new Centralized Sanitary Wastewater Treatment Facility is scheduled to begin in 1994 and should be completed in 1995. This new facility will be built approximately 1 mile south of F-Area on Burma Road. Building the central facility will require clearing approximately 6 acres of planted pines. An 18 mile trunkline/collection system will also be required, using existing transmission line and steam line rights-of-way to the extent possible. This trunkline will be located in the northwest quadrant of the SRS, and will connect the new Centralized Sanitary Wastewater Treatment Facility to A-Area, F-/H-Areas, and C-Area.

Depending on the spent nuclear fuel management alternative chosen, a total of 150 acres of SRS land could be cleared and converted to facilities and infrastructure as a result of spent nuclear fuel management (including an Expended Core Facility), construction of the Savannah River Ecology Laboratory Conference Center, and completion of the Centralized Sanitary Wastewater Treatment Facility. This represents less than 0.1 percent of the undeveloped land on the SRS, and will have minimal cumulative impact on long-term land use locally and regionally.

5.16.2 Socioeconomics

There would be minimal cumulative impacts on the socioeconomic resources of the SRS region from any spent fuel management alternative. The greatest change in employment would occur under the Centralization Alternative, which would include construction and operation of an Expended Core Facility at SRS. Construction of an Expended Core Facility would require an estimated 850 additional employees in the peak year (1999), while operation of the facility would add a maximum of approximate 500 full-time jobs. DOE anticipates that overall employment on the Site will decline during the first 5 years of the spent fuel management period and will stabilize thereafter as the SRS mission changes. Workers who might otherwise lose their jobs could be employed by SRS in spent fuel program activities. Therefore, DOE expects little or no direct increase in employment due to the program. The Site would fill any new jobs from the existing regional labor force.

5.16.3 Air Quality

Table 5-31 compares the cumulative emissions of nonradioactive pollutants from the SRS, including those from the proposed spent nuclear fuel alternatives, to the pertinent regulatory standards.

Table 5-31. Total maximum ground-level concentrations (μg/cubic meter) of criteria and toxic air pollutants at SRS boundary resulting from normal operations and spent nuclear fuel management alternatives.[a,b]

Emissions	Averaging Time	Alternatives 1 through 4		
		Option a Dry Storage	Option b Wet Storage	Option c Processing
Criteria Pollutants				
NO_x	Annual	4 (4%)	4 (4%)	15 (15%)
SO_x	Annual	10 (12%)	10 (12%)	10 (12%)
	24-hours	185.0 (50%)	185.0 (50%)	185.4 (50%)
	3-hours	634 (49%)	634 (49%)	637 (49%)
PM_{10}	Annual	3 (6%)	3 (6%)	3 (6%)
	24-hours	56.0 (37%)	56.0 (37%)	56.4 (37%)
TSP	Annual	11 (17%)	11 (17%)	11 (17%)
Ozone (as VOC)	1-hour	N/A[d]	N/A[d]	N/A[d]
Gaseous fluoride (as HF)	1-month	0.03 (4%)	0.03 (4%)	0.05 (6%)
	1-week	0.15 (9%)	0.15 (9%)	0.25 (16%)
	24-hours	0.31 (11%)	0.31 (11%)	0.51 (18%)
	12-hours	0.62 (17%)	0.62 (17%)	1.02 (28%)
Lead	Annual	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)
CO	8-hours	23.1 (0.2%)	23.1 (0.2%)	27.3 (0.3%)
	1-hour	181 (0.4%)	181 (0.4%)	212 (0.5%)
Toxic Pollutants				
Nitric acid	24-hours	6.7 (5%)	6.7 (5%)	7.7 (6%)
1,1,1-Trichloroethane	24-hours	22 (0.2%)	22 (0.02%)	22 (0.2%)
Benzene	24-hours	31 (21%)	31 (21%)	31 (21%)
Ethanolamine	24-hours	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)
Ethylbenzene	24-hours	0.12 (<0.1%)	0.12 (<0.1%)	0.12 (<0.1%)
Ethylene glycol	24-hours	0.08 (<0.1%)	0.08 (<0.1%)	0.08 (<0.1%)
Formaldehyde	24-hours	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)
Glycol ethers	24-hours	<0.01 N/A	<0.01 N/A	<0.01 N/A
Hexachloronaphthalene	24-hours	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)
Hexane	24-hours	0.07 (<0.1%)	0.07 (<0.1%)	0.11 (<0.1%)
Manganese	24-hours	0.10 (0.4%)	0.10 (0.4%)	0.10 (0.4%)
Methanol	24-hours	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)
Methyl ethyl ketone	24-hours	0.99 (<0.1%)	0.99 (<0.1%)	0.99 (<0.1%)
Methyl isobutyl ketone	24-hours	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)
Methylene chloride	24-hours	1.8 (0.3%)	1.8 (0.3%)	1.82 (0.4%)
Napthalene	24-hours	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)
Phenol	24-hours	0.03 (<0.1%)	0.03 (<0.1%)	0.03 (<0.1%)

Table 5-31. (continued).

Emissions	Averaging Time	Alternatives 1 through 4 Option a Dry Storage	Option b Wet Storage	Option c Processing
Phosphorus	24-hours	<0.001 (<0.2%)	<0.001 (<0.2%)	<0.001 (<0.2%)
Sodium hydroxide	24-hours	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)
Toluene	24-hours	1.6 (8%)	1.6 (8%)	2.0 (10%)
Trichloroethene	24-hours	1.0 (0.3%)	1.0 (0.3%)	1.0 (0.3%)
Vinyl acetate	24-hours	0.02 (<0.1%)	0.02 (<0.1%)	0.02 (<0.1%)
Xylene	24-hours	3.81 (<0.1%)	3.81 (<0.1%)	3.85 (<0.1%)

Emissions	Averaging Time	Alternative 5 - Centralization Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing	Option 5d Ship Out
Criteria Pollutants					
NO_x	Annual	4 (4%)	4 (4%)	15.1 (15%)	4 (4%)
SO_x	Annual	10 (12%)	10 (12%)	10 (12%)	10 (12%)
	24-hours	185.0 (50%)	185.0 (50%)	185.5 (52%)	185.0 (50%)
	3-hours	634.5 (49%)	634.5 (49%)	637.5 (49%)	634 (49%)
PM_{10}	Annual	3 (6%)	3 (6%)	3 (6%)	3 (6%)
	24-hours	56.0 (37%)	56.0 (37%)	56.4 (38%)	56.0 (37%)
TSP	Annual	11 (17%)	11 (17%)	11 (17%)	11 (17%)
Ozone (as VOC)	1-hour	N/A[d]	N/A[d]	N/A[d]	N/A[d]
Gaseous fluoride (as HF)	1-month	0.03 (4%)	0.03 (4%)	0.05 (6%)	0.03 (4%)
	1-week	0.15 (9%)	0.15 (9%)	0.25 (16%)	0.15 (9%)
	24-hours	0.31 (11%)	0.31 (11%)	0.41 (14%)	0.31 (11%)
	12-hours	0.62 (17%)	0.62 (17%)	1.02 (28%)	0.62 (17%)
Lead	Annual	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)
CO	8-hours	24 (0.2%)	24 (0.2%)	28.1 (0.3%)	23.1 (0.2%)
	1-hour	187 (0.5%)	187 (0.5%)	217 (0.5%)	181 (0.4%)
Toxic Pollutants					
Nitric acid	24-hours	6.7 (5%)	6.7 (5%)	7.7 (6%)	6.7 (5%)
1,1,1-Trichloroethane	24-hours	22 (0.2%)	22 (0.02%)	22 (0.2%)	22 (0.2%)
Benzene	24-hours	31 (21%)	31 (21%)	31 (21%)	31 (21%)
Ethanolamine	24-hours	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)
Ethylbenzene	24-hours	0.12 (<0.1%)	0.12 (<0.1%)	0.12 (<0.1%)	0.12 (<0.1%)
Ethylene glycol	24-hours	0.08 (<0.1%)	0.08 (<0.1%)	0.08 (<0.1%)	0.08 (<0.1%)
Formaldehyde	24-hours	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)	<0.01 (<0.1%)
Glycol ethers	24-hours	<0.01 (N/A)	<0.01 (N/A)	<0.01 (N/A)	<0.01 (N/A)
Hexachloronaphthalene	24-hours	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)	<0.01 (<1%)

Table 5-31. (continued).

Emissions	Averaging Time	Alternative 5 - Centralization Option 5a Dry Storage	Option 5b Wet Storage	Option 5c Processing	Option 5d Ship Out
Hexane	24-hours	0.07 (<0.1%)	0.07 (<0.1%)	0.11 (<0.1%)	0.07 (<0.1%)
Manganese	24-hours	0.10 (0.4%)	0.10 (0.4%)	0.10 (0.4%)	0.10 (0.4%)
Methanol	24-hours	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)
Methyl ethyl ketone	24-hours	0.99 (<0.1%)	0.99 (<0.1%)	0.99 (<0.1%)	0.99 (<0.1%)
Methyl isobutyl ketone	24-hours	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)	0.51 (<0.1%)
Methylene chloride	24-hours	1.8 (0.3%)	1.8 (0.3%)	1.82 (0.4%)	1.8 (0.3%)
Napthalene	24-hours	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)
Phenol	24-hours	0.03 (<0.1%)	0.03 (<0.1%)	0.03 (<0.1%)	0.03 (<0.1%)
Phosphorus	24-hours	<0.001 (<0.2%)	<0.001 (<0.2%)	<0.001 (0.2%)	<0.001 (<0.2%)
Sodium hydroxide	24-hours	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)	0.01 (<0.1%)
Toluene	24-hours	1.6 (8%)	1.6 (8%)	2.0 (10%)	1.6 (8%)
Trichloroethene	24-hours	1.0 (0.3%)	1.0 (0.3%)	1.0 (0.3%)	1.0 (0.3%)
Vinyl acetate	24-hours	0.02 (<0.1%)	0.02 (<0.1%)	0.02 (<0.1%)	0.02 (<0.1%)
Xylene	24-hours	3.81 (<0.1%)	3.81 (<0.1%)	3.85 (<0.1%)	3.81 (<0.1%)

a. Source: WSRC (1994a).
b. Numbers in parentheses indicate the percentage of the regulatory standard that each concentration represents.
c. No standard for this chemical.
d. Measurement data currently unavailable.

The values provided are the maximum concentrations that would occur at ground level at the Site boundary. Not all maximum concentrations would occur at the same location.

The data demonstrate that, even with the emissions from the spent nuclear fuel management activities, releases of toxic air pollutants from the SRS would be only a small fraction of the regulatory standards. Therefore, DOE anticipates no cumulative impact.

The releases of some criteria air pollutants by SRS operations would approach regulatory standards. Site sulfur dioxide emissions would reach about 50 percent of both the 24-hour and 3-hour limits under all alternatives. In addition, the emissions of particulates less than 10 microns (PM_{10}) would approach a concentration equal to about 38 percent of the standard. However, the contribution to both these pollutants concentrations made by spent nuclear fuel-related activities would be small, as explained in Section 5.7.

The SRS evaluated the cumulative impact of airborne radioactive releases in terms of cumulative dose to a maximally exposed individual at the Site boundary. Table 5-32 lists the results of this analysis. The highest dose would be 4.7×10^{-1} millirem, which would occur under the processing options of Alternatives 4 and 5. This dose is below the regulatory standard (40 CFR Part 61 Subpart H) of 10 millirem.

Airborne emissions from the two-unit Vogtle Electric Generating Plant (approximately 10 miles southwest of the center of the SRS near Waynesboro, Georgia) were reported to have delivered an MEI total body dose of 1.14×10^{-3} millirem during 1992 (Alison Napier, Halliburton NUS, personal communication with Shan Sumdaram, Georgia Power Company). Since the SRS and Plant Vogtle are essentially proximal to the same 80 kilometer population, the ratio of SRS population and MEI doses was used as an estimator of the population dose due to Plant Vogtle emissions. Using this approach, the population dose attributable to Vogtle was estimated to have been about 8.3×10^{-2} person-rem in 1992. Adding (1) the population dose from Plant Vogtle, (2) the total collective offsite population dose from all SRS activities in 1992 (both air and water source terms), and (3) the highest projected collective dose from spent nuclear fuel management activities (Options 4c and 5c) yields a total cumulative dose of 27.083 person-rem from all SRS sources and Plant Vogtle, which is only 0.3 percent higher than the dose from SRS alone. Note that the doses in Table 5-32 ("Total Collective Dose, Offsite Population") represent the sum of (2) and (3) above.

5.16.4 Water Resources

Approximately 82.1 million gallons per year of Savannah River water would be required for the two most water-intensive options, Option 4f (Regionalization at SRS - Processing) and Option 5c (Centralization - Processing). Because either of these options would probably require construction of an Expended Core Facility, this facility's projected surface water usage of 2.5 million gallons per year was factored into the cumulative impacts analysis. Thus, the two options with the highest surface water usage, both of which would require as much as 84.6 million gallons, represent approximately 0.4 percent of the current (baseline) SRS surface water usage of 20 billion gallons per year (see Table 5-8).

Table 5-32. Annual cumulative health effects to workers and offsite population due to SRS radioactive releases during incident-free operations.

	Worker				Offsite Population			
	Average Individual		Total Collective		Maximally Exposed Individual		Total Collective	
	Dose[a]	Fatal Cancer[b]	Dose[c]	Fatal Cancers[d]	Dose[a]	Fatal Cancer[b]	Dose[c]	Fatal Cancers[d]
Alternative 1 - No Action								
Option 1 Wet Storage	3.2×10^{-1}	1.3×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Alternative 2 - Decentralization								
Option 2a Dry Storage	3.0×10^{-1}	1.2×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 2b Wet Storage	3.2×10^{-1}	1.3×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 2c Processing	3.6×10^{-1}	1.5×10^{-4}	1.6×10^{2}	6.5×10^{-2}	4.4×10^{-4}	2.2×10^{-7}	2.6×10^{1}	1.3×10^{-2}
Alternative 3 - 1992/1993 Planning Basis								
Option 3a Dry Storage	3.0×10^{-1}	1.2×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 3b Wet Storage	3.2×10^{-1}	1.3×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 3c Processing	3.7×10^{-1}	1.5×10^{-4}	1.6×10^{2}	6.6×10^{-2}	4.5×10^{-4}	2.2×10^{-7}	2.6×10^{1}	1.3×10^{-2}
Alternative 4 - Regionalization								
Option 4a Dry Storage	3.0×10^{-1}	1.2×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 4b Wet Storage	3.2×10^{-1}	1.3×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 4c Processing	3.7×10^{-1}	1.5×10^{-4}	1.7×10^{2}	6.8×10^{-2}	4.7×10^{-4}	2.3×10^{-7}	2.7×10^{1}	1.4×10^{-2}
Option 4d Dry Storage	3.2×10^{-1}	1.3×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 4e Wet Storage	3.5×10^{-1}	1.4×10^{-4}	9.4×10^{1}	3.7×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 4f Processing	4.0×10^{-1}	1.6×10^{-4}	1.7×10^{2}	6.8×10^{-2}	4.7×10^{-4}	2.3×10^{-7}	2.6×10^{1}	1.3×10^{-2}
Option 4g Ship Out	$<3.2\times10^{-1}$	$<1.3\times10^{-4}$	$<9.4\times10^{1}$	$<3.7\times10^{-2}$	$<9.0\times10^{-5}$	$<4.5\times10^{-8}$	$<8.9\times10^{0}$	$<4.4\times10^{-3}$

Table 5-32. (continued).

	Worker				Offsite Population			
	Average Individual		Total Collective		Maximally Exposed Individual		Total Collective	
	Dose[a]	Fatal Cancers[b]	Dose[c]	Fatal Cancers[d]	Dose[a]	Fatal Cancers[b]	Dose[c]	Fatal Cancers[d]
Alternative 5								
Option 5a Dry Storage	1.3	5.3×10^{-4}	9.6×10^{1}	3.8×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 5b Wet Storage	1.6	6.4×10^{-4}	9.6×10^{1}	3.8×10^{-2}	9.0×10^{-5}	4.5×10^{-8}	8.9×10^{0}	4.4×10^{-3}
Option 5c Processing	1.6	6.6×10^{-4}	1.7×10^{2}	6.9×10^{-2}	4.7×10^{-4}	2.3×10^{-7}	2.7×10^{1}	1.4×10^{-2}
Option 5d Ship Out	$<3.2 \times 10^{-1}$	$<1.3 \times 10^{-4}$	$<9.4 \times 10^{1}$	$<3.7 \times 10^{-2}$	$<9.0 \times 10^{-5}$	$<4.5 \times 10^{-8}$	$<8.9 \times 10^{0}$	$<4.4 \times 10^{-3}$

a. Dose in rem.
b. Probability of fatal cancer.
c. Dose in person-rem.
d. Incidence of excess fatal cancers.

Operational impacts to surface water quality under any of the spent nuclear fuel management options examined would be minimal. Existing SRS treatment facilities could accommodate all new spent nuclear fuel-related domestic and process wastewater streams. Expected wastewater flows would be well within the design capacities of existing (or planned upgrades of) Site treatment systems. Sanitary wastewater from new spent nuclear fuel facilities would be routed to the new Centralized Sanitary Wastewater Treatment Facility. Liquid radioactive wastes would presumably be sent to the F-/H-Area Effluent Treatment Facility. Treated nonradioactive liquid releases from the new spent nuclear fuel facilities would likely be discharged to Upper Three Runs Creek or Fourmile Branch.

Water quality in the Savannah River downstream of the SRS is adequate to good, with most parameters analyzed showing values below state and Federal Maximum Contaminant Levels or DOE Derived Concentration Guides. Iron, present in soils in the region, is the only constituent of surface waters that routinely exceeds MCLs. Spent nuclear fuel management activities are not expected to result in higher concentrations of iron downstream of the SRS. As noted earlier, in Section 6.0, construction on the new Centralized Sanitary Wastewater Treatment Facility is scheduled to begin in 1994 and should be completed in 1995. The new Centralized Sanitary Wastewater Treatment Facility will replace 14 aging sanitary wastewater facilities with a single state-of-the-art facility which will treat sanitary wastes by an extended aeration-activated sludge process. Chlorine will not be used to treat sanitary wastes in the new facility. Use of non-chemical ultraviolet light disinfection systems will eliminate the use and handling of 32,000 gallons of sodium hypochlorite and 59,000 gallons of sodium sulfite per year. Eliminating these chemicals has essentially eliminated the potential for toxic chemical releases from the wastewater treatment process.

Operation of the new Centralized Sanitary Wastewater Treatment Facility and closure of the old A-, B-, S-Area, and Naval Fuel sanitary wastewater facilities would also eliminate wastewater discharges to Upper Three Runs Creek, the stream on the SRS least degraded by past operations. Treated effluent from the new Centralized Sanitary Wastewater Treatment Facility will discharge to Fourmile Branch. Overall stream quality in Fourmile Branch is expected to improve because the effluent from the new facility will be cleaner than the effluent from the old package plants in C-, F-, and H-Areas that presently discharge to Fourmile Branch. As a result, the cumulative effect of the new spent nuclear fuel management facilities (any alternative considered) and new Centralized Sanitary Wastewater Treatment Facility will probably be a net improvement in water quality in two SRS streams, Upper Three Runs Creek and Fourmile Branch, and may result in better water quality downstream in the Savannah River as well.

Sanitary wastewater from the new Consolidated Incineration Facility will be routed to the new Centralized Sanitary Wastewater Treatment Facility; there will be no direct process wastewater drains to the environment. Liquid wastes will be collected in storage tanks and periodically trucked to a permitted hazardous/mixed waste treatment and disposal facility. Sanitary wastes from the new Savannah River Ecology Laboratory Conference Center will be piped to a septic tank-drain field system and would not impact surface water in the area.

Sanitary wastes produced during construction of the Expended Core Facility would be treated through the use of portable chemical toilets or through an existing wastewater treatment facility. Depending on the location chosen by DOE and the Navy for the new Expended Core Facility, sanitary wastes from operation of the ECF would either be treated in an existing wastewater treatment facility (most likely the new Centralized Sanitary Wastewater Facility) or a new treatment facility designed to handle the facility's wastewater capacity. No process wastes from operation of the Expended Core Facility will be discharged to the environment.

5.16.5 Occupational and Public Health and Safety

Table 5-32 summarizes the cumulative health effects of incident-free SRS operations, including those projected for the spent nuclear fuel alternatives. The table lists potential cancer facilities for workers and the public due to radiological exposures to airborne and waterborne releases from the Site. In addition, the table provides the (airborne) dose to the hypothetical maximally exposed individual in the offsite population. As explained in Chapter 5, the evaluation used 1992 as the baseline year for normal operations, because it is the last year for which the SRS has complete information. DOE believes that this year gives a realistic depiction of current operational releases of radionuclides. The assessment added the estimated releases from each spent fuel alternative to this baseline to determine the cumulative impacts listed in Table 5-32.

5.16.6 Waste Management

The analysis of cumulative impacts of SRS waste management activities takes as its starting point the assumption that waste generation under the No Action Alternative represents the baseline condition for the entire Savannah River Site. Waste generation levels associated with the other proposed spent nuclear fuel management alternatives (see Table 5-18) thus represent positive and negative deviations from this baseline. Cumulative effects of the proposed spent nuclear fuel alternatives on the volume of low-level waste, transuranic waste, and high-level waste produced under each of the proposed alternatives are presented in Table 5-30. Environmental restoration and cleanup activities, which are expected to become an increasingly important part of the DOE mission in the future, have not been factored into this analysis. These activities are expected to produce large quantities of radioactive, hazardous, and mixed wastes; however, these environmental restoration activities will be the focus of another environmental impact statement.

5.17 Unavoidable Adverse Environmental Impacts

The construction and operation of facilities related to any of the five alternatives at the Savannah River Site (SRS) would result in some adverse impacts to the environment. Changes in project design and other methods of mitigation could eliminate, avoid, or reduce most of these to minimal levels. The following paragraphs identify adverse impacts that mitigation could not reduce to minimal levels or avoid altogether.

The generation of some fugitive dust during construction would be unavoidable, but would be controlled by water and dust suppressants. This would occur under Alternatives 2 to 5, but greatest generation of dust would occur under Alternative 5 (excluding the offsite shipping option). Similarly, construction activities would result in some minor, yet unavoidable, noise impacts from heavy equipment, generators, and vehicles.

The maximum loss of habitat would involve the conversion of 70 to 100 acres (0.28 to 0.4 square kilometer) of managed pine forest to industrial land use; this would occur under Alternative 5 if DOE moved all spent nuclear fuel to the SRS.

The amount of radioactivity that normal operation of the spent nuclear fuel facilities would release under four of the five alternatives (Alternatives 1 to 4) would be a small fraction of the 1992 operational releases at the SRS and would be well below applicable regulatory standards.

For the alternative having the most impact (Alternative 5 - Centralization), DOE has calculated that the maximum probability for latent fatal cancer for the maximally exposed member of the public would be about 3 times higher than that calculated for 1992 at the SRS. For latent fatal cancer incidence in the offsite population, this comparison indicates an increase of about 2 times, but the number of cancers calculated is less than one.

The only socioeconomic impacts of the proposed spent nuclear fuel management facilities would be temporary increases in employment and expenditures in the region of influence during the construction phase. These would be unavoidable beneficial impacts.

15.18 Relationship Between Short-Term Use of the Environment and the Maintenance and Enhancement of Long-Term Productivity

Implementation of any of the proposed alternatives would result in some short-term resource demands (e.g., fuel, construction materials, and labor) and would, under certain alternatives (notably the Centralization Alternative), reduce the natural productivity of a relatively small tract of land (less than .07 percent of total SRS area) currently committed to timber production. Depending upon the precise location selected for facility development, a small amount of marginal-to-good wildlife habitat (see Chapters 4.9 and 5.9) would also be lost when the area is cleared, graded, and committed to facilities and supporting infrastructure. However, these short-term resource losses and land-use restrictions provide a basis for improved productivity and utility over the long term at the SRS because consolidating all spent nuclear fuel at a few onsite locations would free for other uses those locations presently committed to spent fuel management. On a national scale, the interim management plan described in this EIS would have the same impact of making locations throughout the DOE complex available for other long-term uses.

5.19 Irreversible and Irretrievable Commitments of Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of facilities related to the spent nuclear fuel alternatives would involve materials that could not be recovered or recycled or that would be consumed or reduced to unrecoverable forms. The construction and operation of spent nuclear fuel facilities at the SRS would consume irretrievable amounts of electrical energy, fuel, concrete, sand, gravel, and miscellaneous chemicals. Other resources used in construction would probably not be recoverable. These would include finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery. Construction and operation of facilities for spent nuclear fuel management would also require the withdrawal of water from surface- and groundwater sources, but most of this water would return to onsite surface streams or the Savannah River after use and treatment.

The Centralization alternative (Option 5c - Processing) would consume the greatest amount of electricity of any of the alternatives, about 110,400 megawatt-hours. The Processing option (excluding Option 4c, Regionalization by fuel type) would have the highest requirements for coal to produce steam, approximately 2,580 metric tons (2,843 tons) annually. The Centralization alternative (except Option 5d where all spent fuel would be shipped off the site) would involve the greatest irretrievable consumption of other resources, such as construction materials, chemicals, gases, and operating supplies. However, this demand would not constitute a permanent drain on local resources or involve any material that is in short supply in the region.

5.20 Mitigation

This section summarizes the measures that DOE could use to mitigate impacts to the environment caused by spent nuclear fuel management activities at the SRS. DOE would determine the extent to which any mitigation would be necessary and the selection of which measures would be implemented during a detailed site-specific National Environmental Policy Act review tiered from this

Programmatic EIS. Consequently, the following sections in this chapter address mitigation in general terms and describe typical measures that the SRS could implement. In addition, the analyses described in this appendix indicates that the environmental consequences of spent fuel management would be minimal in most environmental media. Accordingly, no mitigation would be necessary.

5.20.1 Pollution Prevention

DOE is committed to comply with Executive Order 12856, "Federal Compliance with Right-to-Know Laws and Pollution Prevention Requirements"; Executive Order 12780, "Federal Acquisition, Recycling and Waste Prevention"; and applicable DOE Orders and Guidance Documents in planning and implementing pollution prevention at the SRS. The pollution prevention program at the Site was initiated in 1990 as a waste minimization program. Currently, the program consists of four major initiatives: solid waste minimization; source reduction and recycling of wastewater discharges; source reduction of air emissions; and potential procurement of products manufactured from recycled materials. Since 1991, the waste of all types generated at the SRS has decreased, with greatest reductions in hazardous and mixed wastes. These reductions are attributable primarily to material substitutions.

All spent fuel management activities at the SRS would be subject to the Site pollution prevention program. Implementation of the program plan will mitigate waste generated by these activities.

5.20.2 Socioeconomics

Spent nuclear fuel activities would have minimal impact on the socioeconomic environment in the region of influence because most employees would be drawn from the existing site workforce. The minor impacts of in-migrating construction workers could be mitigated by DOE keeping local communities and county planning agencies informed as to scheduling of construction activities.

5.20.3 Cultural Resources

A Programmatic Memorandum of Understanding (SRARP 1989) between the DOE Savannah River Operations Office, the South Carolina State Historic Preservation Office, and the Advisory Council on Historic Preservation, ratified on August 24, 1990, is the instrument for the management of cultural resources at the SRS. DOE uses this memorandum to identify cultural resources and develop mitigation plans for affected resources in consultation with the State Historic Preservation Officer.

DOE would comply with the terms of the memorandum for all activities needed to support the spent nuclear fuel management activities at the Site. For example, DOE would survey sites prior to disturbance and could mitigate any potentially significant resources encountered through avoidance or removal. Any artifacts encountered would be protected from further disturbance and the elements until removed.

DOE conducted an investigation of Native American concerns over religious rights in the Central Savannah River Valley in conjunction with studies in 1991 related to a New Production Reactor. During this study, three Native American groups expressed concern over sites and items of religious significance on the SRS (see Section 4.4.2). DOE has included these organizations on its environmental mailing list, solicits their comments on National Environmental Policy Act actions of the Site, and sends them documents about SRS environmental activities, including those related to these SNF management considerations. These Native American groups would be consulted on any actions that may follow subsequent site-specific environmental reviews.

5.20.4 Geo' :gy

DOE expects that there would be no impacts to geologic resources at the SRS under any alternative evaluated in this Draft EIS. Potential soil erosion in areas of ground disturbance would be minimized through sound engineering practices such as implementing controls for stormwater runoff (e.g., sediment barriers), slope stability (e.g., rip-rap placement), and wind erosion (e.g., covering soil stockpiles). Re-landscaping would minimize soil loss after construction was completed. These mitigation measures would be included in a site-specific Storm Water Pollution Prevention Plan that the SRS would prepare prior to initiating any construction.

5.20.5 Air Resources

DOE would meet applicable standards and permit limits for all radiological and non-radiological releases to the atmosphere. In addition, the SRS would follow the DOE policy of maintaining radiological emissions to levels "as low as reasonably achievable" (ALARA). ALARA is an approach to radiation protection to control or manage exposures (both individual and collective) and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit. ALARA is not a dose limit, but rather a process that has as its objectives the attainment of dose levels as far below the applicable limits as practicable.

5.20.6 Water Resources

DOE would minimize the potential for adverse impacts on surface water during construction through the implementation of a stormwater pollution prevention plan that details controls for erosion and sedimentation. The plan would also establish measures for prevention of spills of fuel and chemicals and for rapid containment and cleanup.

DOE could mitigate water usage during both construction and operation of facilities by instituting water conservation measures such as instructing workers in water conservation (e.g., turn off hoses when not in use), installing flow restrictors, and using self-closing hose nozzles.

5.20.7 Ecological Resources

DOE does not anticipate that any of the spent fuel alternatives would impact any wetlands on the Site. In any case, DOE and SRS policy is to achieve "no net loss" of wetlands. Pursuant to this goal, DOE has issued a guidance document, *Information for Mitigation of Wetland Impacts at the Savannah River Site* (DOE 1992), for project planners that puts forth a practical approach to wetlands protection that begins with avoidance of impacts (if possible), moves to minimization of impacts (if avoidance is impossible), and requires compensatory mitigation (wetlands restoration, creation, or acquisition) in the event that impacts cannot be avoided.

The analysis in this Draft EIS indicates that there are no threatened and endangered species or sensitive habitats in the areas considered as representative of potential sites for spent nuclear fuel activities at the SRS. However, DOE would perform site-specific predevelopment surveys to ensure that development of new facilities would not impact any of these biological resources.

5.20.8 Noise

DOE anticipates that noise impacts both on and off the Site would be minimal. Mitigation measures would include proper maintenance of exhaust mufflers on construction equipment and trucks.

5.20.9 Traffic and Transportation

DOE has a system of onsite buses operating at the SRS. The Site would evaluate the need for upgrades or changes in service that might be required for the spent nuclear fuel management activities and would make changes, as necessary.

DOE would manage changes in traffic volume or patterns during construction through such measures as designating routes for construction vehicles, providing workers with safety reminders, and upgrading onsite police traffic patrols, if necessary.

5.20.10 Occupational and Public Health and Safety

The DOE program for maintaining radiological emissions to levels "as low as reasonably achievable" (ALARA) described in Section 5.20.5 above will minimize any impacts to workers and the public due to atmospheric releases. Likewise, the Site Pollution Prevention Plan and emergency preparedness measures will enhance safety both on and off the Site.

5.20.11 Utilities and Support Services

The utilities and support services at the SRS are sufficient to meet the requirements of any of the alternatives for the spent fuel management at the Site. Impacts on these services would be minimal. No mitigation measures would be required.

5.20.12 Accidents

The SRS has in place emergency action plans that would be activated in the case of an accident. These plans contain both onsite provisions (e.g., evacuation plans, response teams, medical and fire response, training and drills, communications equipment) and offsite arrangements (e.g., response plans for medical and fire agencies, coordination with local and state agencies, communication plans). The SRS plans would be updated to include any new facilities or activities related to spent nuclear fuel management that would involve the Site. The execution of the plans in response to an accident would mitigate adverse effects both on the Site and in the surrounding areas.

ATTACHMENT A: ACCIDENT ANALYSIS

A.1 Accident Evaluation Methodologies and Assumptions

The potential for facility accidents and the magnitude of their consequences is an important factor in the evaluation of the spent nuclear fuel alternatives addressed in this EIS. There are two health risk issues:

- Would accidents at any of the Savannah River Site (SRS) facilities that the U.S. Department of Energy (DOE) could build for spent nuclear fuel management activities pose unacceptable health risks to workers or the general public?

- Could alternative locations or facilities for the spent nuclear fuel alternatives provide smaller public or worker health risks? Smaller risks could arise from such factors as greater isolation of the facility from the public, a reduced frequency of such external accident initiators as seismic events or aircraft crashes, reduced inventory, and process differences.

Guidance for the implementation of Council on Environmental Quality (CEQ) regulations (40 CFR Part 1502.22), as amended (50 CFR Part 15618), requires the evaluation of impacts that would have a low probability of occurrence but high consequences if they did occur; this EIS, therefore, addresses facility accidents to the extent feasible.

A.1.1 Radiological Accident Evaluation Methodology

The alternatives considered in this EIS provide an opportunity to incorporate new features and technology in new facilities, processes, and operations that would minimize the possibility of undue risk to the health and safety of plant workers and the public. Modifications and upgrades could mitigate accident consequences from existing facilities or reduce the likelihood of occurrence.

Under normal circumstances, DOE would develop accident scenarios and calculate accident consequences using safety analyses, mitigation features, and design details on proposed facility designs. However, the preliminary design information for the proposed facilities that is available during the preparation of this EIS does not contain sufficient detail to permit quantitative safety analyses. Therefore, for each spent nuclear fuel alternative, DOE has evaluated the existing and proposed facilities for the type of radiological accidents it has determined to be reasonably foreseeable.

The radiological accident types fell into four categories: (1) fuel damage, (2) material releases, (3) nuclear criticalities, and (4) liquid spills or discharges. For each accident type, DOE determined reference accidents by examining DOE-approved safety analysis reports (SARs) and other appropriate documentation (e.g., previous EISs). In addition, DOE considered accidents from adjacent facilities for their possible impacts related to spent nuclear fuel. DOE extracted the overall frequency for each reference accident from the appropriate source, rather than attempting to calculate individual frequencies for all possible initiators; that is, DOE did not use the specific probability of a certain magnitude earthquake to determine the frequency of a criticality or spill, given the occurrence of the earthquake. If multiple initiators could lead to one of the reference accidents, or the combined frequency of the initiators could lead to one of the reference accidents, DOE used the combined frequency of the initiators, generally providing conservative results. For example, the Receiving Basin for Offsite Fuel has a number of potential release initiators that could result in an uncontrolled criticality, as listed in Table A-1. As listed, a number of incidents, all of which have their own assigned frequencies, can contribute to the initiation of an uncontrolled criticality.

Table A-1. Potential release initiators at the Receiving Basin for Offsite Fuel.

Natural Phenomena	External Events	Operations Induced Events	Criticality
Temperature Extreme	Aircraft Crash	Fuel Cutting	Fuel Bundling Error
Snow	Helicopter Crash	Spill at Hose Rack	Cask Loading Error
Rain	Surface Vehicle Crash	Fuel Rupture in Storage	Fuel Identification Problem
Lightning		Fire and Explosion	Fuel Movement Error
Tornado		Fuel Near Basin Surface	Dropped Fuel
Earthquake		Spills and Leaks	Cranes or Hoist Collapse
Meteorite Impact		Resin Regeneration Facility Waste to Cell	Cask Immersion Error

This evaluation results in qualitative comparisons for proposed facilities based on the assumption that the facility function is similar to one already analyzed. In addition, an identical set of initiators is not considered in each safety analysis report for existing SRS facilities because these reports were prepared over several years in accordance with requirements in effect at the time. Section A.2

includes a comparison of the similarities of possible facilities to an existing facility, the basis for the selection of reference accidents, and several tables containing data to support a comparison of point estimates of risk.

The qualitative comparison supports the NEPA process, in that the decisionmaker can assess the relative risk from each alternative at SRS and other sites.

A.1.1.1 Notable Accident Initiators. While there are many different types of accident initiators of various frequencies that could lead to an accident, three notable initiators - criticalities, earthquakes, and aircraft crashes - require additional discussion due to the public's perception of the importance of these initiators and the public's familiarity with these types of initiators.

Because there has never been an uncontrolled criticality accident at the SRS, DOE must use historic experience related to the initiators to estimate the frequency for a criticality incident in the Receiving Basin for Offsite Fuel. Storage basins for spent nuclear fuel have excellent safety histories. From 1945 through 1980, there were 40 known criticality accidents worldwide, none of which occurred in a fuel storage facility. From 1975 to 1980, there were, conservatively, 160 reactors with storage basins in operation around the world, and no criticality incidents occurred. Therefore, DOE assumes that the upper frequency limit for a criticality event is 3.1×10^{-3} per year (Du Pont 1983). This figure is applicable to the extent that the storage basins and the operations performed in them are similar to those of the Receiving Basin for Offsite Fuel. However, the frequency for a processing criticality event was determined through a detailed fault tree analysis, as referenced in the safety analysis report, to be an overall calculated limit of 1.4×10^{-4} per year. This value accounts for the implementation of new administrative controls or equipment.

The SRS is in an area that has a relatively low seismic frequency. Based on three centuries of recorded seismic activity, an earthquake with a Richter magnitude greater than 6.0, which corresponds to a Modified Mercalli Intensity Scale (MMI) of VII, would not be likely at the SRS. The design-basis earthquake for the SRS is a MMI VIII event with a corresponding horizontal peak ground acceleration of 0.2*g*. Based on current technology, as applied in various probabilistic evaluations of the seismic hazard in the SRS region, the 0.2*g* peak ground acceleration can be associated with a 2×10^{-4} annual probability of exceedance (5,000-year return period). There are four scenarios for the

Receiving Basin for Offsite Fuel to which an earthquake of intensity MMI VIII or greater might contribute:

- Deformation of the storage racks leading to a criticality incident.
- Derailment of the 100-ton (91-metric-ton) crane into the storage basin with the deformation of the storage rack leading to criticality.
- Damage to the basin walls leading to the release of contaminated basin water to the subsoil.
- Rupture of a waste tank or pipe in the Resin Regeneration Facility leading to the release of contaminated liquids.

An aircraft crash into a spent nuclear fuel facility is of concern because it could result in a radioactive release of materials from the stored spent nuclear fuel. Appendix D contains an aircraft crash probability analysis based on the examination of large civilian and military aircraft crossing the airspace within a 10-mile (16-kilometer) radius of the SRS. It does not include the crash probability of general aviation aircraft because aircraft of this type generally do not possess sufficient mass or attain sufficiently high velocities to produce a serious radiological threat in the event that they crashed into an area containing spent nuclear fuel. The analysis did not evaluate crash probabilities with a likelihood of occurrence of less than 10^{-7} per year because they would not significantly contribute to the risk. This was the case for spent nuclear fuel facilities located at the SRS.

A.1.1.2 Use of DOE-Approved Safety Documents. The NEPA guidance issued by the DOE Office of NEPA Oversight, dated May 1993, recommends that accident impact analyses "reference Safety Assessments and Safety Analysis Reports, if available." This guidance was the primary basis used to develop the approach used in the accident analysis section of this EIS. This Appendix uses several relevant safety analysis reports as well as a previously published EIS. Safety analysis reports are the primary source of information on credible accidents with the potential to cause a release of hazardous materials. These reports are required for all reactors and nuclear materials facilities with operations that potentially pose a significant hazard to onsite personnel, offsite populations, or the environment. The referenced safety analysis reports and EIS approval/draft submittal dates encompass a range from 1983 to 1993. The 1983 safety analysis report was supplemented by a 1993 addendum; the next oldest safety analysis report was approved in 1988.

A.1.2 Chemical Hazard Evaluation Methodology

This analysis reviewed the appropriate safety analyses to assess the degree to which they addressed chemical accidents. It found that each of the safety analyses addressed chemical hazards in a qualitative manner. To provide a quantitative discussion of chemical hazards, the analysis evaluated a separate risk assessment (WSRC 1993c) for the storage risk of offsite research reactor fuel in the Receiving Basin for Offsite Fuel to determine a bounding chemical accident. The analysis determined chemical inventories (see Section A.3) for the existing spent nuclear fuel facilities at the SRS using the "Savannah River Site Tier Two Emergency and Hazardous Chemical Inventory Report" (WSRC 1994a) to determine the facilities total chemical inventory. This chemical inventory was further screened using the EPA's "List of Lists" (EPA 1990).

A.1.3 SRS Emergency Plan

The SRS emergency plan (WSRC 1993b) defines appropriate response measures for the management of emergencies (e.g., accidents) involving the Site. It incorporates into one document a description of the entire process designed to respond to and mitigate the potential consequences of an accident. Emergencies that could cause activation of all or portions of this plan include:

- Events (operational, transportation, etc.) with the potential to cause releases above allowable limits of hazardous materials.

- Events such as fires, explosions, tornadoes, hurricanes, earthquakes, dam failures, etc., that affect or could affect safety systems designed to protect site and offsite populations and the environment.

- Events such as bomb threats, hostage situations, etc., that reduce the security posture of the Site.

- Events created by proximity to other facilities, such as the Vogtle Electric Generating Plant, a commercial nuclear powerplant located across the Savannah River from the Site.

For radiological emergencies, protective actions in this plan are designed to keep onsite and offsite exposures As Low As Reasonably Achievable (ALARA). This is accomplished by minimizing time spent in the vicinity of the hazard, keeping as far from the hazard as possible, and taking advantage of available shielding. Protective actions that could be used on the Site in the event of an emergency include remaining indoors, sheltering, evacuation, and relocation. For events that cause an actual or projected radiological release, appropriate protective actions for on- and offsite populations have been determined based on trigger points called Protective Action Guides (PAGs).

A.1.4 General Assumptions

This assessment applied the following key assumptions to examine existing accident analyses and to relate these analyses to the spent nuclear fuel alternatives.

- When a referenced accident scenario is used for a possible new facility, DOE would build the new facility close to an existing referenced facility performing a similar function, resulting in consequences and health effects similar to the existing facilities analyzed. The exception could be the proposed Expended Core Facility which Appendix D analyzes separately.

- For existing facilities to be modified, portions of the facility to be decommissioned, or new facilities to be added, potential accident initiators resulting from construction and nearby activities would be bounded by the referenced accident scenarios.

- Type 2 High Enriched Uranium fuel, the dominant type currently in storage or process at the SRS, would provide a reference source term for other fuel types (i.e., Mark 22 fuel).

- Spent nuclear fuel acceptance criteria would specify that all fuel must be capable of indefinite suspension in air with no melting.

- The total frequency of an event (e.g., criticality) could be used to determine point estimates of risk, regardless of the type or specific frequencies of the individual contributing initiators.

- Adjustment (scaling) factors could be applied to reflect a best engineering judgment in terms of relative risk between the various alternatives.

- The point estimate of risk for a given accident scenario would be representative in that it could, for the purposes of this programmatic EIS, represent a similar accident scenario at new facilities that perform similar functions.

- Reference accidents would be attributed to a facility based on its function (e.g., fuel canning or dry material storage) regardless of whether the facility currently exists, is undergoing design, or is in the conceptual design phase.

- Possible new facilities would be designed to pose no greater risk to the workers and public than existing facilities with similar functions.

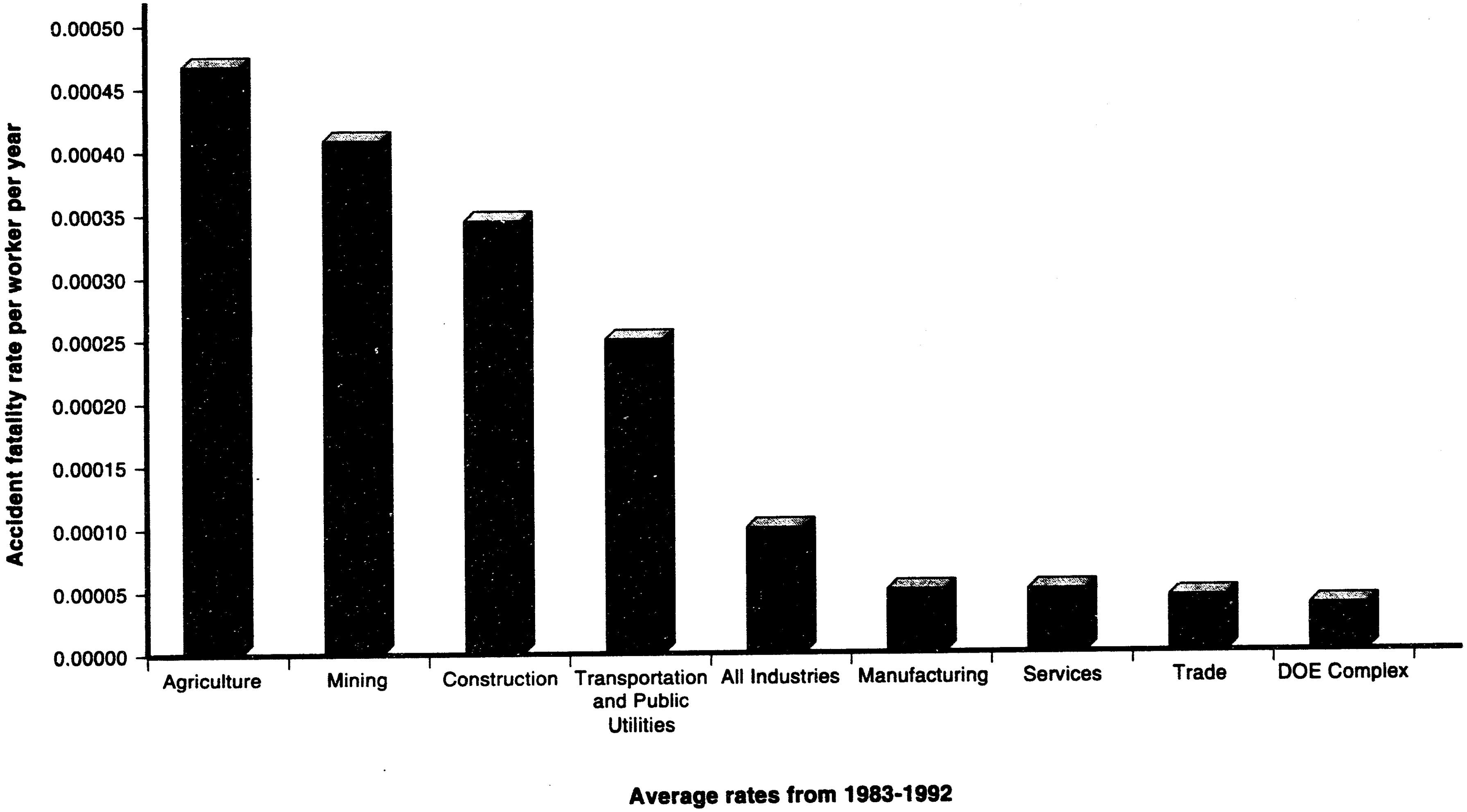
This evaluation takes no credit for the upgraded design requirements for the proposed facilities. Such facilities should have improved reliability or mitigative features and, therefore, would reduce the aggregate frequency of accidents. Therefore, the application of values from existing safety analysis reports would provide conservative results. In addition, the evaluation makes no attempt to discriminate among similar existing facilities that might have slightly different frequencies of occurrence or source terms (i.e., an FB-Line event frequency was applied to HB-Line and other processing facilities).

For most accidents, the evaluation did not quantify consequences for workers. The safety analysis reports from which information was extracted for the reference accidents were written before the issuance of DOE Order 5480.23 (DOE 1992); previous applicable Orders did not require the inclusion of worker doses. The historic record indicates that DOE facilities have an enviable safety record. Figure A-1 compares the rate of worker fatalities in the DOE complex (DOE 1993) to national average rates compiled by the National Safety Council for various industry groups (NSC 1993). Because the DOE worker accident fatality rate compares favorably to rates from such industry groups as agriculture and construction and is slightly less than trade and services group rates, the absence of quantitative data regarding accident impacts to radiological workers should not impede the decisionmaking process. The discussion presented in Volume 1 adequately addresses the impacts for close-in workers (i.e., those directly involved in the activity or near the accident source) at the SRS.

A.1.4.1 Receptor Group Assumptions. To ensure comparative results, the evaluation assessed the measures of impacts among four receptor groups:

- Worker. An individual located 100 meters (328 feet) in the worst sector of a facility location where the release occurs.

- Colocated Worker. An individual located 640 meters (2,100 feet) in the worst sector of a facility location where the release occurs.



Source: National Safety Council, 1993; DOE 1993

SFIG 0A01

Figure A-1. Comparison of fatality rates among workers in various industry groups.

- Maximally Exposed Offsite Individual (MEI). A hypothetical resident located at the nearest Site boundary from the facility location where the release occurs.

- Offsite Population to 80 Kilometers. The collective sum of individuals located within an 80-kilometer (50-mile) radius of the SRS.

As noted above, the worker is 100 meters (328 feet) from the facility where the accident occurs. This is because information quantifying accident impacts (i.e., dose and health effects) to workers at less than 100 meters from an accidental release of radionuclides is unavailable. For each of the accident scenarios considered in Appendix C of this EIS, there is some risk of worker injury or death at distances closer than 100 meters. Furthermore, the safety analyses from which this evaluation extracted information for the accident scenarios often did not include any discussions on worker impacts as a result of potential accidents. DOE Orders published before DOE 5480.23 (DOE 1992) did not require the inclusion of worker doses. However, Section A.2.6.2 includes a qualitative discussion regarding accident impacts for the worker at less than 100 meters (328 feet) for each of the radiological accident scenarios.

A.1.4.2 Code Assumptions. DOE's application of the AXAIR and AXAIR89Q (a validated version) dose estimation models is acceptable for projecting health effects from accidents at SRS and comparing the results to results from other similar codes (RSAC-5 and GENII) used at other sites. AXAIR is a Gaussian model based on the methodology outlined in NRC Regulatory Guide 1.145 (NRC 1983). AXAIR contains a meteorological data file specific to SRS that provides conservative calculated doses for the radiological consequences of atmospheric releases. AXAIR and AXAIR89Q include the following specific functions:

- Performs both environmental transport and radiation dosimetry calculations

- Bases environmental transfer models on NRC Reg Guide 1.145 guidelines

- Includes exposure pathways for inhalation of radionuclides and gamma radiation from the radioactive plume

- Calculates gamma shine doses using nonuniform Gaussian model

- Uses worst sector and 99.5-percentile meteorology

Doses calculated with this code should bound the radiological consequences for atmospheric releases postulated.

A.1.4.3 Criticality Assumptions. An estimate of the consequences of a criticality incident requires an estimate of the number of fissions that might occur. While U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.34 specifies 1×10^{19} fissions as the upper tenth of incidence experience, the SRS analyses are based on mean values, to the extent possible, for all incidents. Criticality incidents have produced from 10^{14} to 4×10^{19} fissions with a mean of 2×10^{18} fissions for incidents involving fissile solutions and a mean of 5×10^{17} fissions for incidents involving solids. As a consequence, two accident scenarios (Table A-2) address criticality - the wet pool criticality scenario and the processing criticality scenario. For the wet pool criticality scenario, the mean value for solid systems (5×10^{17}) is assumed to apply to the source term used to determine the accident consequences, while the processing criticality scenario assumes that the mean value for a solution (2×10^{18}) was applied to the source term to determine accident consequences.

A.2 Radiological Accident Scenarios

A.2.1 Selection of Reference Accidents

To support the examination of both existing and proposed facilities, this evaluation considered a spectrum of potential accident types. To develop a meaningful spectrum of potential accidents, the evaluation posed the following question:

> "What could be done to spent nuclear fuel that would result in a radiological consequence to the receptor groups?"

In determining the answer to this question, the following four general types of events emerged: (1) fuel damage, (2) material releases, (3) criticalities, and (4) liquid spills or discharges. A review of applicable safety analysis reports for the SRS facilities that the spent nuclear fuel alternatives would be likely to affect generated more than 20 accidents involving the transport, receipt, processing, and storage of spent nuclear fuel. A consolidation and subsequent "binning" of these accidents for each accident type reflects an appropriate range of case-specific reference accidents.

Table A-2. Reference radiological accidents considered for spent nuclear fuel activities.

	Name and Reference	Reference for Source Term/Dose	Comparative Likelihood/Frequency
A1.	Fuel Assembly Breach Reference Accident: RBOF fuel cutting	Tables 1-3 DPSTSA-200-10-3, Addendum 1	1.6×10^{-1} per year
A2.	Material Release (Processing) Reference Accident: FB-Line release	Tables 5-30 and A-4 DPSTSA-200-10-9	5 per year (total from pg. 5-2, tables 5.4, 5.5, and 5.8)
A3.	Material Release (Dry Vault) Reference Accident: PSF release	Table 5-9 DPSTSA-200-10-19	1.4×10^{-3} per year
A4.	Material Release (Adjacent Facility) Reference Accident: Release of Waste Tank Activity to Cell	Tables 1-3 DPSTSA-200-10-3, Addendum 1	2.4×10^{-3} per year
A5.	Criticality in Water Reference Accident: RBOF criticality	Tables 1-3 DPSTSA-200-10-3, Addendum 1	3.1×10^{-3} per year
A6.	Criticality During Processing Reference Accident: FB-Line	Tables 5-27, 5-28, and 5-29 DPSTSA-200-10-9	1.4×10^{-4} per year
A7.	Spill/Liquid Discharge (External) Reference Accident: ROEIS direct discharge from disassembly basin	Table 4-8 DOE/EIS-0147	1 event/Rx life $\frac{1}{126 \text{ yr}} = 7.9 \times 10^{-3}$ per year
A8.	Spill/Liquid Discharge (Internal) Reference Accident: RBOF hose rack spill	Tables 1-3 DPSTSA-200-10-3, Addendum 1	1.1×10^{-1} per year

The fuel damage event (type 1 accident) considered was physical damage or breaching of a fuel assembly. Three material (type 2 accidents) releases were considered; they represent releases that could occur during processing from medium energetic events, those that could occur during dry storage of special nuclear materials, and those that could occur from an adjacent facility. Criticality (type 3 accidents) can have different dose impacts and can occur with different frequencies, depending on the physical or chemical characteristics of the material and the surroundings. Two criticality events - in water and during processing - represent these accident scenarios. The evaluation considered a dry criticality accident scenario bounded by the wet pool criticality in terms of frequency and bounded by the processing criticality accident in terms of number of fissions assumed. Two liquid discharges and spills (type 4 accidents) were considered - inadvertent discharges of pool or basin water assumed to contain tritium and other radioactive constituents from the fuel in the pool (external spill), and spills of slightly contaminated liquids inside a facility during fuel handling, spraying, or cask unloading, (internal spill).

These eight typical accidents form the set of accidents for the selection of a reference accident. Each type has been assigned an alphanumeric designator, which is listed below and used throughout this document:

- Type 1 - Fuel damage

  A1 - Fuel assembly breach

- Type 2 - Material releases

  A2 - Processing release
  A3 - Dry vault release
  A4 - Adjacent facility release

- Type 3 - Criticalities

  A5 - Criticality in water
  A6 - Criticality during processing

- Type 4 - Liquid discharges and spills

  A7 - External spill/liquid discharge
  A8 - Internal spill/liquid discharge

A second review of the safety analyses and the original list of accidents confirmed that each specific accident considered in DOE-approved safety analyses could be represented or bounded by one of the eight "generic" accidents (i.e., a fire could result in material release or an earthquake could result in criticality or liquid release). The use of this approach with documented total frequencies avoids the need for unique identification of all initiating precursor events or their specific probabilities.

A.2.1.1 Externally Initiated Accidents. The accident analysis section of this EIS considered accident scenarios from external events or adjacent facilities and their potential impacts on direct spent nuclear fuel activities and facilities. Three significant sources of externally induced accident mechanisms were identified as potentially applicable to these facilities and activities: aircraft crashes, adjacent fires, and adjacent explosions. As discussed above, an aircraft crash scenario is not a credible event within the probability scope of this EIS. For the most part, a fire or explosion in a facility adjacent to the spent nuclear fuel facilities described in Figure 3-2 would not have a significant impact on spent nuclear fuel facilities. However, the screening process determined that a fire and explosion in the Resin Regeneration Facility, located immediately adjacent to the Receiving Basin for Offsite Fuel, could result in the airborne release to the shielded cell and should be included for completeness.

A.2.1.2 Nearby Industrial or Military Facility Accidents. Within a 40-kilometer (25-mile) radius of the SRS, there are approximately 120 industrial facilities with 25 or more employees (DOE 1990). Four of these facilities are within a 16-kilometer (10-mile) radius of the SRS. Other than those on the SRS, the only major storage facilities within a 40-kilometer radius are the facilities at Chem-Nuclear Systems, Inc., Vogtle Electric Generating Station, and a cluster of natural gas storage tanks near Beech Island. The facilities within a 16-kilometer radius of the SRS boundary are still at least 10 kilometers (6 miles) from the nearest spent nuclear fuel facility, and thus present negligible risk to spent nuclear fuel activities.

A.2.1.3 Common Cause Accident. DOE considered accident scenarios based on a common cause accident during the screening process. However, considering the fact that there were no common cause accident analyses addressed in available SRS safety documentation, this evaluation did not include the cumulative impacts of simultaneous accidents. The SRS does maintain emergency plans that would provide protective actions and mitigate consequences that could occur during a common cause accident scenario.

A.2.1.4 Accidents Resulting from Terrorism. DOE considered accident scenarios based on a terrorist attack or an act of sabotage during the screening process and concluded that any accident resulting from such initiators would be bounded by or similar to the accident scenarios already considered.

A.2.2 Reference Accident Descriptions

DOE established a reference accident for each of the eight generic or typical accidents. The following paragraphs outline the basis for selection of each reference accident by scenario. A reference accident was included if it is analyzed in an SRS safety analysis report that has been approved by the DOE or submitted to DOE for approval as part of the safety basis authorizing operation of a facility, and if the facility is to be utilized as, or is similar in function to, one of the facilities included in the five alternatives and their subordinate cases. For example, the analysis assumed that the Receiving Basin for Offsite Fuel was representative of any spent nuclear fuel wet storage pool. If an accident could occur in any pool, the analysis selected a reference scenario from the Receiving Basin for Offsite Fuel Safety Analysis Report as the reference accident, as listed in Table A-2. The following paragraphs provide the basis for each selection.

- **A1. Fuel Assembly Breach** - Physical damage to an assembly could occur from dropping, objects falling onto the assembly, or cutting into the fuel part of an assembly. The Receiving Basin for Offsite Fuel Safety Analysis Report (WSRC 1993a) Addendum contains a current analysis of a "fuel cutting accident." The inert, non-uranium-containing extremities of some spent nuclear fuel elements are cut off (cropped) in the repackaging basin before the bundling of the elements. The spent nuclear fuel could be inadvertently cut, causing a release of airborne or high water activity to the work area. Because of the metallic nature of SRS fuel, only a very small fraction of the gases generated in an assembly would be released to the basin water in an accident. Consistent with the safety analysis report, fuel cooled for 90 days is used in the source term for this accident. With foreign research reactor spent nuclear fuel elements, the release of fission product gases would be less than with the Mark 22 fuel assemblies previously considered. The physics of the release of gases from research reactor fuel is similar to SRS fuel because the fuel is constructed in a similar manner. Spent nuclear fuels that could release more fission gases than a Mark 22 fuel assembly would require an Unreviewed Safety Question analysis before the SRS could accept them in the Receiving Basin for Offsite Fuel. Air monitors in this area would warn personnel in the event of an airborne release. The fuel cutting operation involves only one fuel element at a time. This is representative for all cutting and dropping accidents because cracking the cladding would release less than cutting into the fuel itself.

- **A2. Material Release (Processing)** - Medium energetic events in the various stages of processing (e.g., dissolution, separation, and evaporation) were the dominant contributors to material releases. A medium energetic event is defined as one that will cause penetration of the primary confinement barrier, and will cause materials to bypass the second confinement barrier for a short period of time. Medium energetic events not related to nuclear criticality such as uncontrolled chemical reactions, fires, or external impact events can result in the dispersal of radioactive materials. This evaluation assumes that the fractions of the plutonium volatized and transported are the same as those applied to the dispersal of the nonvolatile fission products of a criticality. Because these events were analyzed for the types of fuel processed at the SRS and because FB-Line releases result in greater impacts than those from HB-Line, the collective result (i.e., the total frequency of medium energetic events, not just the highest release event) for FB-Line medium energetic events was selected as the reference accident.

- **A3. Material Release (Dry Vault)** - Accident types A1 and A2 cover material releases from fuel handling and processing. In addition, DOE considered a reference accident for vault-type storage. The Plutonium Storage Facility (PSF) Safety Analysis Report (Du Pont 1989) analyzed three medium energetic events (shipping container failure, criticality, and impact-type events) and an earthquake. As discussed above, medium energetic events are accidents that result in release of material from the primary container and have sufficient energy to penetrate the secondary confinement barriers for a short period of time. That report contains a total frequency of these four initiating events and provides one release value. Because the SRS has no long-term spent nuclear fuel dry storage facilities, this evaluation assumes that the Plutonium Storage Facility vault is representative of dry storage facilities, as are the activities and precursor events. A material release from any medium energetic event in the Plutonium Storage Facility was selected as the reference accident for nonprocessing material releases.

- **A4. Material Release (Adjacent Facility)** - For completeness, DOE considered a reference accident from a facility immediately adjacent to the Receiving Basin for Offsite Fuel (WSRC 1993a). This scenario includes a fire and explosion at the Resin Regeneration Facility in waste tank EP 38 during which the coolant of a received cask, when discharged to the waste tank, results in a flammable or explosive concentration of vapors in the tank. Rupture of the tank by an explosion could release airborne activity to the shielded cell if the accident occurred during one of the projected 150 times per year when regeneration of the portable columns takes place. While a fire and explosion have not occurred in waste tank EP 38, one fire and pressure surge did occur when a shipping cask was being vented. The spent nuclear fuel remained intact and radionuclides were not released. The incident has been attributed to the ignition of a mixture of hydrogen, oxygen, and air emanating from the cask and created by reaction of hot aluminum fuel with water left in the cask by the shipper.

- **A5. Criticality in Water** - This scenario assumes that a wet pool storage facility is the most likely to have a criticality in water. The Receiving Basin for Offsite Fuel provides the capability for underwater receipt, handling, and storage of spent nuclear fuel. Primary radiation shielding is provided by the water over the spent nuclear fuel. A safety analysis report determined frequency and results from many initiating events that could lead to criticality. The following activities could ultimately lead to a criticality incident: Fuel Bundling, Cask Loading, Fuel Identification and Manifest Problems, Fuel Movement, Dropped Fuel, Fuel Near Basin, Cask Immersion, and Cranes and Hoist. These events are representative for any wet storage pool.

- **A6. Criticality During Processing** - As noted in the discussion for accident type A2, FB-Line events are representative for SRS processing facilities. The analysis considered the total of the frequencies for criticality initiators for all processing stages, which would, therefore, be conservative because not all processing stages would necessarily be involved in a new facility and not all stages would necessarily occur simultaneously.

- **A7. Spill/Liquid Discharge (External)** - The reference accident selected for this type of event is the direct discharge of the K-Reactor disassembly basin to a stream. The EIS on Continued Operation of K-, L-, and P-Reactors (DOE 1990) considers several alternatives to the use of a seepage basin for routine discharges of tritiated disassembly-basin water. The direct-discharge dose impacts were the highest of the alternatives considered. The selection of the direct-discharge event is conservative for existing or possible new facilities in F- and H-Areas because no free-flowing surface streams would be near a discharge point. The use of the source term from the reactor disassembly basin is considered to be conservative for the spent nuclear fuel storage pools. Although the disassembly basin has water-circulating systems to control radioactivity, chemistry, clarity, and temperature, these processes are inferior to those used in the Receiving Basin for Offsite Fuel. Activated corrosion products, particulate activities, tritium, and other radioactive contaminants (e.g., Cs-137) are in the basin water. The use of direct discharge to a stream is conservative because the scenario considers all contaminants to be deposited, assuming no decay time.

- **A8. Spill/Liquid Discharge (Internal)** - DOE considered a second reference accident for contaminated liquids spills or discharges was considered by DOE to ensure the appropriate onsite impacts. The discharge discussed for accident type A7 would be external to the building and would have no measurable worker impact component because the reference accident occurred outside the facility. The Receiving Basin for Offsite Fuel hose rack spill was selected as the reference accident because it is representative of small, unplanned, but relatively frequent spills in a storage facility and could impact the worker. Minor releases of contaminated water could occur at the hose rack platform during the handling of portable deionizers for the reactor areas.

A.2.3 Source Term and Frequency Determinations

Table A-2 lists source term references from existing documents approved by DOE or submitted by Westinghouse Savannah River Company to DOE for approval for each selected reference accident. The same references nominally prescribed the frequency of accidents or initiating events. If it was not directly available, the frequency was derived from information already contained in the appropriate safety analysis report or EIS (e.g., if only a risk estimate and a dose were listed, the frequency was derived by dividing the risk by the dose). These frequencies fall into ranges associated with abnormal events (more frequent than 1×10^{-3} per year), design-basis accidents (1×10^{-3} per year to 1×10^{-6} per year), or beyond-design-basis accidents (less than 1×10^{-6} per year to 10^{-7} per year).

This document does not analyze beyond-design-basis accidents or accidents with frequencies of less than 1.0×10^{-6} explicitly because the accident analysis source material (DOE-approved safety analysis reports) considers these accidents to be incredible events. Beyond-design-basis accidents, such as an airplane crash-induced criticality, have no different consequences (i.e., number of fissions) than the criticality estimated to occur with a frequency of 3.1×10^{-3} per year. Because of the use of aggregate frequencies in some cases, the contribution to overall risk from 1.0×10^{-7} per year events is negligible, and the higher frequency initiators dominate the point estimate of risk. Some initiating or precursor event frequencies from the safety analysis reports are at 10^{-7} per year or lower; thus, these reports in fact consider events beyond the 10^{-6} frequencies.

Frequencies for reference accidents were determined as follows:

- **A1. Fuel Assembly Breach** - The frequency for this reference accident was obtained from DPSTSA-200-10-3, *Receiving Basin for Offsite Fuel (RBOF)*, Addendum 1, Tables 1-5, which lists the frequency as 1.6×10^{-1} per year (WSRC 1993a).

- **A2. Material Release (Processing)** - The frequency for this reference accident was obtained from DPSTSA-200-10-9, *Safety Analysis - 200 Area, Savannah River Plant, FB-Line Operations*, April 1988, Tables 5-4, 5-5, and 5-8, from which a combined frequency of five events per year was determined (Du Pont 1988).

- **A3. Material Release (Dry Vault)** - The frequency for this reference accident was obtained from DPSTSA-200-10-19, *Final Safety Analysis Report - 200 Area, Savannah River Site Separations Area Operations, Building 221F, B-Line, Plutonium Storage Facility,* July 1989, Tabl: 5-9, which lists the frequency as 1.4 x 10^{-3} per year (Du Pont 1989).

- **A4. Material Release (Adjacent Facility)** - The frequency for this reference accident was obtained from DPSTSA-200-10-3, Receiving Basin for Offsite Fuel (RBOF), Addendum 1, Tables 1-5, which lists the frequency as 2.4 x 10^{-3} per year (WSRC 1993a).

- **A5. Criticality in Water** - The frequency for this reference accident was obtained from DPSTSA-200-10-3, *Receiving Basin for Offsite Fuel (RBOF),* Addendum 1, Tables 1-5, which lists the frequency as 3.1 x 10^{-3} per year (WSRC 1993a).

- **A6. Criticality During Processing** - The frequency for this reference accident was obtained from DPSTSA-200-10-9, *Safety Analysis - 200 Area, Savannah River Plant, FB-Line Operations,* April 1988, page 5-34, which lists a frequency of 1.4 x 10^{-3} per year.

- **A7. Spill/Liquid Discharge (External)** - The frequency for this reference accident was derived from DOE/EIS-0147, *Continued Operation of K-, L-, and P-Reactors,* December 1990, which described a direct discharge from a disassembly basin. The event was divided over the 126-year operational span of the SRS production reactors for a frequency of 7.9 x 10^{-3} per year (DOE 1990).

- **A8. Spill/Liquid Discharge (Internal)** - The frequency for this reference accident was obtained from DPSTSA-200-10-3, *Receiving Basin for Offsite Fuel (RBOF),* Addendum 1, Tables 1 - 3, which lists the frequency as 1.1 x 10^{-1} per year for a representative spill at a hose rack (WSRC 1993a).

A.2.4 Applicability of Accidents to Facilities

This evaluation reviewed Section 1 of the reference document *Technical Data Summary Supporting the Spent Nuclear Fuel Environmental Impact Statement* (WSRC 1994b) to develop a matrix of the selected radiological accidents to the facilities (modules) being considered for the various alternatives and cases. For proposed new facilities, the analysis used best engineering judgment to extrapolate from appropriate accident scenarios based on the descriptions provided in the reference document. Table A-3 lists the connection of facilities to accident scenarios. For example, the Examination and Characterization Facility (module B) identifies a potential accident scenario, A1 (as defined in Table A-2), that should be considered when this facility is utilized to support any case.

Table A-3. Applicable accidents and facilities.

Facility	Module[a]	Accidents
Spent Fuel Receiving, Cask Handling and Fuel Unloading	A	A1
Examination and Characterization	B	A1
Naval Reactor Spent Fuel Examination and Characterization	C	A1, A5, A7, A8
Spent Fuel Repackaging	D	A1, A5, A7, A8
Canister Loading	E	A1, A7, A8
Interim Dry Storage	F	A1, A3
Interim Spent Fuel Storage Pool	G	A1, A5, A7, A8
F-Canyon/F-Area Separations	H, I	A1, A2, A3, A6
H-Canyon/H-Area Separations	J, K, L	A1, A2, A3, A6
Reactor Disassembly Basins	M	A1, A5, A7
Receiving Basin for Offsite Fuels	N	A1, A4, A5, A7, A8

a. As defined in WSRC (1994b).

A.2.5 Facilities and Reference Accidents Associated with each Alternative Case

Table A-4 links alternatives, specific cases, supporting facilities (modules), and accident scenarios. This table identifies the facilities that could be required to support each alternative by specific case. The combined associated accident scenarios for each facility provide the accident spectrum associated with the specific cases for each alternative.

A.2.6 Impacts from Radioactive Release Accidents

This section provides a quantitative discussion of potential consequences to the receptor groups identified in Section A.1.4.1. It also provides a qualitative discussion on potential health effects and consequences for workers at less than 100 meters (328 feet) for each of the potential accident scenarios.

Table A-4. Spent nuclear fuel facilities and accident spectrum by alternatives.

Alternative	Modules[a]	Accidents
1. NO ACTION		
Option 1 - Wet Storage	M, N	A1, A4, A5, A7, A8
2. DECENTRALIZATION - allows development of new facilities R&D, limited fuel transportation close by, and desirable safety upgrades (beyond essential).		
Option 2a - Dry Storage	B, D, E, F, G, M, N	A1, A3, A4, A5, A7, A8
Option 2b - Wet Storage	B, D, E, G, M, N	A1, A4, A5, A7, A8
Option 2c - Processing	G, H, I, J, K, L, M, N	A1, A2, A3, A4, A5, A6, A7, A8
3. PLANNING BASIS - like the usual no-action alternative (i.e., continue status quo, fuel stored wet stays wet, new SNF expected still arrives, previous planned mods/upgrades or new facilities all right.		
Option 3a - Dry Storage	B, D, E, F, G, M, N	A1, A3, A4, A5, A7, A8
Option 3b - Wet Storage	B, D, E, G, M, N	A1, A4, A5, A7, A8
Option 3c - Processing	G, H, I, J, K, L, M, N	A1, A2, A3, A4, A5, A6, A7, A8
4. REGIONALIZATION - redistribute SRS SNF to keep only AL-clad, upgrades, and new as needed.		
Option 4a - Dry Storage	A, B, D, E, F, G, M, N	A1, A3, A4, A5, A7, A8
Option 4b - Wet Storage	A, B, D, E, G, M, N	A1, A4, A5, A7, A8
Option 4c - Processing	A, G, H, I, J, K, L, M, N	A1, A2, A3, A4, A5, A6, A7, A8
Option 4d - Dry Storage	A, B, C, D, E, F, G, M, N	A1, A3, A4, A5, A7, A8
Option 4e - Wet Storage	A, B, C, D, E, G, M, N	A1, A4, A5, A7, A8
Option 4f - Processing	A, C, G, H, I, J, K, L, M, N	A1, A2, A3, A4, A5, A6, A7, A8
Option 4g - Ship Out	M, N	A1, A4, A5, A7, A8
5. CENTRALIZATION - all SNF comes to SRS, extensive new facilities, or all SNF shipped out.		
Option 5a - Dry Storage	A, B, C, D, E, F, G, H, M, N	A1, A3, A4, A5, A7, A8
Option 5b - Wet Storage	A, B, C, D, E, G, M, N	A1, A4, A5, A7, A8
Option 5c - Processing	A, C, G, H, I, J, K, L, M, N	A1, A2, A3, A4, A5, A6, A7, A8
Option 5d - Ship Out	M, N	A1, A4, A5, A7, A8

a. Source: Westinghouse (1994b).

A.2.6.1 Radioactive Release Accidents and Consequences for Spent Nuclear Fuel Alternatives. Table A-5 summarizes the information in Tables A-2 through A-4 and provides individual consequences (doses) based on accident type for each case. The table lists consequences for the four receptor groups as follows: Maximum Offsite Individual Dose, the Population to 80 kilometers (50 miles) Dose, the Worker Dose, and the Colocated Worker Dose.

Table A-5. Radioactive release accidents and consequences for spent nuclear fuel alternatives.

Description		Accident	Accident frequency (per year)	Maximally offsite individual dose (rem)	Population to 80 kilometers dose (person-rem)	Worker dose (person-rem)	Colocated worker dose (person-rem)
			1. NO ACTION				
Option 1 Wet Storage	A1	Fuel Assembly Breach	$1.6x10^{-1}$	$2.0x10^{-3}$	$1.7x10^{1}$	(a)	$1.2x10^{-2}$
	A4	Material Release (adjacent facility)	$2.4x10^{-3}$	$6.0x10^{-3}$	$5.0x10^{1}$	(a)	$5.0x10^{-2}$
	A5	Criticality in Water	$3.1x10^{-3}$	$3.0x10^{-3}$	$8.8x10^{0}$	(a)	$1.4x10^{-1}$
	A7	Spill/Liquid Discharge (external)	$7.9x10^{-3}$	$1.7x10^{-2}$	$2.7x10^{-2}$	(a)	(a)
	A8	Spill/Liquid Discharge (internal)	$1.1x10^{-1}$	$2.4x10^{-10}$	$2.0x10^{-6}$	(a)	$2.0x10^{-11}$
			2. DECENTRALIZATION				
Option 2a Dry Storage	A1	Fuel Assembly Breach	$1.6x10^{-1}$	$2.0x10^{-3}$	$1.7x10^{1}$	(a)	$1.2x10^{-2}$
	A3	Material Release (dry vault)	$1.4x10^{-3}$	$2.1x10^{-6}$	$6.9x10^{-3}$	(a)	(a)
	A4	Material Release (adjacent facility)	$2.4x10^{-3}$	$6.0x10^{-3}$	$5.0x10^{1}$	(a)	$5.0x10^{-2}$
	A5	Criticality in Water	$3.1x10^{-3}$	$3.0x10^{-3}$	$8.8x10^{0}$	(a)	$1.4x10^{-1}$
	A7	Spill/Liquid Discharge (external)	$7.9x10^{-3}$	$1.7x10^{-2}$	$2.7x10^{-2}$	(a)	(a)
	A8	Spill/Liquid Discharge (internal)	$1.1x10^{-1}$	$2.4x10^{-10}$	$2.0x10^{-6}$	(a)	$2.0x10^{-11}$
Option 2b Wet Storage	A1	Fuel Assembly Breach	$1.6x10^{-1}$	$2.0x10^{-3}$	$1.7x10^{1}$	(a)	$1.2x10^{-2}$
	A4	Material Release (adjacent facility)	$2.4x10^{-3}$	$6.0x10^{-3}$	$5.0x10^{1}$	(a)	$5.0x10^{-2}$
	A5	Criticality in Water	$3.1x10^{-3}$	$3.0x10^{-3}$	$8.8x10^{0}$	(a)	$1.4x10^{-1}$
	A7	Spill/Liquid Discharge (external)	$7.9x10^{-3}$	$1.7x10^{-2}$	$2.7x10^{-2}$	(a)	(a)
	A8	Spill/Liquid Discharge (internal)	$1.1x10^{-1}$	$2.4x10^{-10}$	$2.0x10^{-6}$	(a)	$2.0x10^{-11}$
Option 2c Processing	A1	Fuel Assembly Breach	$1.6x10^{-1}$	$2.0x10^{-3}$	$1.7x10^{1}$	(a)	$1.2x10^{-2}$
	A2	Material Release (processing)	$5.0x10^{0}$	$1.7x10^{-7}$	$1.3x10^{-3}$	$2.7x10^{1}$	(a)
	A3	Material Release (dry vault)	$1.4x10^{-3}$	$2.1x10^{-6}$	$6.9x10^{-3}$	(a)	(a)
	A4	Material Release (adjacent facility)	$2.4x10^{-3}$	$6.0x10^{-3}$	$5.0x10^{1}$	(a)	$5.0x10^{-2}$
	A5	Criticality in Water	$3.1x10^{-3}$	$3.0x10^{-3}$	$8.8x10^{0}$	(a)	$1.4x10^{-1}$
	A6	Criticality in Processing	$1.4x10^{-4}$	$2.6x10^{-3}$	$2.9x10^{0}$	$2.4x10^{3}$	(a)

Table A-5. (continued).

Description		Accident	Accident frequency (per year)	Maximally offsite individual dose (rem)	Population to 80 kilometers dose (person-rem)	Worker dose (person-rem)	Colocated worker dose (person-rem)
				1. NO ACTION			
	A7	Spill/Liquid Discharge (external)	7.9×10^{-3}	1.7×10^{-2}	2.7×10^{-2}	(a)	(a)
	A8	Spill/Liquid Discharge (internal)	1.1×10^{-1}	2.4×10^{-10}	2.0×10^{-6}	(a)	2.0×10^{-11}
				3. PLANNING BASIS			
Option 3a Dry Storage				Same as Option 2a for Decentralization			
Option 3b Wet Storage				Same as Option 2b for Decentralization			
Option 3c Processing				Same as Option 2c for Decentralization			
				4. REGIONALIZATION			
Option 4a and 4d Dry Storage				Same as Option 2a for Decentralization			
Option 4b and 4e Wet Storage				Same as Option 2b for Decentralization			
Option 4c and 4f Processing				Same as Option 2c for Decentralization			
Option 4g Ship Out				Same as Alternative 1, No Action			
				5. CENTRALIZATION			
Option 5a Dry Storage				Same as Option 2a for Decentralization			
Option 5b Wet Storage				Same as Option 2b for Decentralization			
Option 5c Processing				Same as Option 2c for Decentralization			
Option 5d Ship Out				Same as Alternative 1, No Action			

a. The safety analysis reports from which information was extracted for these accidents were written before the issuance of DOE Orders 5480.23 (DOE 1992); previous orders did not require the inclusion of worker doses.

A.2.6.2 Impacts to Workers at Less than 100 Meters from Radiological Releases.

This section provides a qualitative discussion addressing the impacts due to potential radiological accident scenarios to workers at less than 100 meters (328 feet) involved in SRS spent nuclear fuel management. While worker fatalities may result from release initiators (i.e., plane crashes, seismic event, crane failure, etc.) and not as a direct consequence of a radiation release, this discussion considers only the radiological impacts of an accident, should it occur.

- **A1. Fuel Assembly Breach** - No fatalities to workers would be expected from radiological consequences because the release of the source term would be underwater. Attenuation by the water would occur for most products, but the release of noble gases would cause a direct radiation exposure to workers in the area. However, because of the high metallic content of SRS spent nuclear fuel, only a very small fraction of the gases generated in an assembly would be released to the basin water. Air monitors in the area would warn personnel in the event of an airborne release. Timely evacuation would prevent substantial radiation exposures.

- **A2. Material Release (Processing)** - No fatalities to workers would be likely from radiological consequences. This scenario assumes that a material release would be distributed into the volume of the smallest room for each unit of operation. Further, it assumes that the operator would be able to exit the room in 30 seconds (Du Pont 1988). This scenario presumes that the fractions of the plutonium volatized and transported are the same as those applied to the dispersal of the nonvolatile fission products of a criticality. Based on these assumptions, radiological exposure to the worker could occur.

- **A3. Material Release (Dry Vault)** - No fatalities to workers would be likely from radiological consequences. Medium energetic events resulting in the release of radioactive material from the Plutonium Storage Facility vault can result in the dispersal of radioactive materials. For these events, the radioactive material present would bypass the containment and disperse, but would result in a dose well below the lethal level. This assumes that a material release would be distributed into the volume of the smallest room for each unit of operation. It is further assumed that the operator is able to exit the room in 30 seconds (Du Pont 1989). This scenario presumes that the fractions of the plutonium volatized and transported are the same as those applied to the dispersal of the nonvolatile fission products of a criticality. Based on these assumptions, radiological exposure to the worker could occur.

- **A4. Material Release (Adjacent Facility)** - No fatalities to workers would be likely from radiological consequences. The rupture of a waste tank by an explosion could release airborne activity to the shielded cell if the accident occurred during one of the projected 150 times per year when regeneration of the portable columns took place (WSRC 1993d). Although some radiological exposure to the worker could occur, the risk to the worker from the initiating fire and explosion would predominate. Air monitors in the area would warn personnel in the event of an airborne release. Timely evacuation would prevent substantial radiation exposures.

- **A5. Criticality in Water** - No fatalities to workers would be likely from radiological consequences. The use of casks and the underwater handling of spent nuclear fuel greatly reduces the possibility of over-exposure of workers to radiation. The approximately 3 meters (10 feet) of water that covers all fuel provides an attenuation factor of 10^5 for intense gamma radiation and provides protection from direct radiation, even in the event of a criticality. However, a small chance of direct radiation exposure could result due to a floating fuel element or a fuel element inadvertently being raised too high. Strategically located radiation monitors reduce even this probability by alerting workers and sounding an evacuation alarm.

- **A6. Criticality During Processing** - The radiation field generated by a criticality incident could lead to fatalities among workers at the FB-Line facility. As discussed in Section A.2.2, FB-Line inadvertent criticality events are bounding for F- and H-Area spent fuel management processing facilities. This is assumed because workers involved in the FB-Line activities are in close proximity to plutonium metal. Of the 74 personnel that could be present during normal operations, 56 are expected to be within areas which the safety analysis report (Du Pont 1988) identifies as potential criticality accident locations. The shielding due to the concrete floors and walls, the distance between personnel, and the specific nature of the event reduce personnel dose so that only nearby personnel on the floor where the accident occurred would potentially receive a fatal dose. In the event of a criticality accident, DOE estimates that up to 4 deaths could occur, and as many as 50 other workers could receive non-fatal levels of direct radiation.

- **A7. Spill/Liquid Discharge (External)** - No fatalities to workers would be likely from radiological consequences because drainage of the large amount of water in a water pool is expected to take several days, which provides sufficient time for workers to leave the area.

- **A8. Spill/Liquid Discharge (Internal)** - No fatalities to workers would be likely from radiological consequences. Minor releases of contaminated water have occurred at the Receiving Basin for Offsite Fuel hose rack platform during the handling of portable deionizers from the reactor areas. One such release was the result of an operator attempting to correct a small leak on a pressurized portable deionizer and was subsequently sprayed with contaminated water resulting in a radioactive exposure. A spill at the hose rack is not expected to release more than 378.5-liters (100 gallons) of contaminated water.

A.2.7 Point Estimates of Risk

Table A-6 lists the point estimate of risk for each reference accident considered for two receptors. The point estimate of risk is the product of frequency (in occurrences per year) and the number of potential latent fatal cancers. The number of potential latent fatal cancers is the product of dose (in rem for the individual or person-rem for the population) and the ICRP 60 risk factors (4.0×10^{-4} latent fatal cancer per rem for the worker or 5.0×10^{-4} latent fatal cancer per rem for the general public). These point estimates were used to determine the relative risk for each case and to determine the accident that becomes dominant if DOE retires specific facilities during the total period under consideration. For example, all alternatives begin with the immediate storage of spent nuclear fuel in wet pools; however, for the alternative considering interim dry storage, the accident dominating risk will change as the configuration of facilities utilized changes and as spent nuclear fuel or special nuclear material is placed in and remains in interim storage rather than being handled.

Table A-6. Point Estimates of Risk for Reference Accident Scenarios.

			Potential Fatal Cancers[a]		Point of Estimate of Risk[b]	
Accident Scenario	Descriptions	Frequency (per year)	Maximally Exposed Individual	Population to 80 kilometers	Maximally Exposed Individual	Population to 80 kilometers
A1	Fuel Assembly Breach	1.6×10^{-1}	1.0×10^{-6}	8.5×10^{-3}	1.6×10^{-7}	1.4×10^{-3}
A2	Material Release (processing)	5.0×10^{0}	8.5×10^{-11}	6.5×10^{-7}	4.3×10^{-10}	3.3×10^{-6}
A3	Material Release (dry vault)	1.4×10^{-3}	1.1×10^{-9}	3.5×10^{-6}	1.5×10^{-12}	4.9×10^{-9}
A4	Material Release (adjacent facility)	2.4×10^{-3}	3.0×10^{-6}	2.5×10^{-2}	7.2×10^{-9}	6.0×10^{-5}
A5	Criticality in Water	3.1×10^{-3}	1.5×10^{-6}	4.4×10^{-3}	4.7×10^{-9}	1.4×10^{-5}
A6	Criticality in Processing	1.4×10^{-4}	1.3×10^{-6}	1.5×10^{-3}	1.8×10^{-10}	2.1×10^{-7}
A7	Spill/Liquid Discharge (external)	7.9×10^{-3}	8.5×10^{-6}	1.4×10^{-5}	6.7×10^{-8}	1.1×10^{-7}
A8	Spill/Liquid Discharge (internal)	1.1×10^{-1}	1.2×10^{-13}	1.0×10^{-9}	1.3×10^{-14}	1.1×10^{-10}

a. ICRP 60 risk factor (5.0×10^{-4}) latent fatal cancer per rem was used to determine potential latent fatal cancers.
b. Units for point estimates of risk are given in potential fatal cancers per year.

A.2.8 Fuel Transition Staging Risk

Table A-7 facilitates the examination of the dominant reference accident during the fuel handling, processing, and storage stages. The use of stages enabled a realistic comparison of risk over the evaluated period. For example, when all fuel has been unloaded, characterized, canned, and put into an interim storage position, consideration of fuel handling events is no longer meaningful.

Table A-7. Dominant risks based on fuel transition stages.

Fuel/Material Stage	Maximally Exposed Individual Risk	Population to 80 Kilometers Risk
Wet storage	1.6×10^{-7} potential fatal cancer/yr based on accident scenario A1.	1.4×10^{-3} potential fatal cancer/yr based on accident scenario A1.
Dry storage	1.5×10^{-12} potential fatal cancers/yr based on accident scenario A3.	4.9×10^{-9} potential fatal cancers/yr based on accident scenario A3.
Processing (fuel "in-process" by DOE definition)	4.3×10^{-10} potential fatal cancer/yr based on accident scenario A2.	3.3×10^{-6} potential fatal cancer/yr based on accident scenario A2.

A.2.9 Adjustment Factors for Comparison Between Alternatives

The accident scenarios described in this document (i.e., Appendix C) differ only slightly between the various alternatives. The scenarios do not account for variations in spent nuclear fuel shipments (including onsite operational transfers) and spent nuclear fuel storage inventories across the alternatives. To provide a realistic comparison across alternatives, DOE developed factors to adjust frequencies or consequences, depending on the specific circumstances of each alternative. This section describes the methodology and justification used to develop adjustment (scaling) factors for a relative comparison of adjusted point estimates of risk for each alternative on a case-by-case basis.

A.2.9.1 Classification of SRS Accident Scenarios for Applicability to Adjustment Factors. This evaluation screened the SRS accident scenarios to determine which adjustment factor categories were applicable. Table A-8 lists the classification of the different SRS accident scenarios. These adjustment categories are as follows:

- Frequency sensitive due to spent nuclear fuel handling
- Frequency sensitive due to spent nuclear fuel inventories
- Consequence sensitive due to spent nuclear fuel inventories

Table A-8. Adjustment factor classification of SRS accidents.

Accident Scenarios	Accident Description	Frequency Sensitive (Handling)	Frequency Sensitive (Inventory)	Consequence Sensitive (Inventory)
A1	Fuel Assembly Breach	X		
A2	Material Release (Processing)		X	
A3	Material Release (Dry Vault)			X
A4	Material Release (Adjacent Facility)	X		
A5	Criticality in Water	X		
A6	Criticality during Processing		X	
A7	Spill/Liquid Discharge (External)			X
A8	Spill/Liquid Discharge (Internal)			X

The following paragraphs provide the basis for each category selection:

- **A1. Fuel Assembly Breach** - The major initiator for this accident is the mishandling of a fuel assembly. For this reason, the accident frequency for this accident is adjusted to account for the annual number of fuel handling events. The amount of material involved in this accident is limited by the amount of damage that will occur due to the mishandling of a fuel assembly. Therefore, the bounding consequences of this accident are constant and independent of the amount of material available.

- **A2. Material Release (Processing)** - The probability that a release could occur during processing depends on the amount of material that will be processed. Therefore, the accident frequency for this accident is adjusted based on the spent nuclear fuel inventory. Because a maximum amount of material can be processed at any one time, the bounding consequences of this accident are independent of the amount of material on the site.

- **A3. Material Release (Dry Vault)** - The major contributor to the probability of occurrence for this release was external initiators that did not involve material handling. This supports using the same frequency for each alternative. The consequences of this accident are proportional to the amount of material available for release. Therefore, the bounding consequences for this accident are based on the amount of material to be stored.

- **A4. Material Release (Adjacent Facility)** - The initiator for this accident involves the discharge of coolant from a cask into a waste tank. The frequency of occurrence for this accident depends on the number of casks received; therefore, the frequency is adjusted to account for the annual number of fuel shipments.

- **A5. Criticality in Water** - The probability of occurrence of this accident was determined by considering the probability of occurrence of several initiating events. Many of these initiating events involved a criticality due to the mishandling of fuel. Therefore, the frequency for this accident is adjusted to account for the annual number of fuel handling events. The magnitude of the criticality accident is not a function of the amount of material available because the criticality is a highly unlikely, localized event. The consequences for this accident are not adjusted to account for the amount of material available.

- **A6. Criticality During Processing** - The probability that a criticality could occur during processing depends on the amount of material that will be processed. Therefore, the frequency for this accident is adjusted based on the spent nuclear fuel inventory. The magnitude of the criticality accident is not a function of the amount of material available because the criticality is a highly unlikely, localized event. The consequences for this accident are not adjusted to account for the amount of material available.

- **A7. Spill/Liquid Discharge (External)** - The major contributor to the probability of occurrence for this release was external initiators that did not involve material handling. This supports using the same frequency for each alternative. The consequences depend on the amount of fuel in the basin because an increase in the amount of fuel will increase the source term in the basin water. Therefore, the bounding consequences are adjusted for the amount of fuel to be stored.

- **A8. Spill/Liquid Discharge (Internal)** - The major contributor to the probability of occurrence for this release was external initiators that did not involve material handling. This supports using the same frequency for each alternative. The consequences depend on the amount of fuel in the basin because an increase in the amount of fuel will increase the source term in the basin water. For this reason the bounding consequences are adjusted for the amount of fuel to be stored.

A.2.9.2 Methodology for Determination of Onsite Shipping Frequencies. This section discusses the methodology for determining the onsite shipping frequencies of spent nuclear fuel on a case-by-case basis for each alternative. The annual frequency of handling accidents will vary in direct proportion to the annual number of handling events. However, the consequences of the accident will not vary as a result of spent nuclear fuel handling activities because the amount of material involved in each handling event does not vary. This evaluation assumes that onsite shipments of spent nuclear fuel are near-term shipments, averaged over 5 years. Table A-9 provides a breakdown of current spent nuclear fuel inventories at SRS facilities.

Table A-9. Spent nuclear fuel inventories.[a]

Facility	Number of Aluminum Assemblies[b]	Number of Aluminum Slugs (Buckets[c])	Number of Nonaluminum-Clad Assemblies	Number of Aluminum-Clad Assembly Shipments	Number of Aluminum-Clad Bucket Shipments	Number of Nonaluminum-Clad Assembly Shipments
Receiving Basin for Offsite Fuel (RBOF)	234	107 (2)	261	20	1	22
K-Reactor Basin	1,783	349 (7)	0	149	3	0
L-Reactor Basin	861	13,840 (256)	0	72	86	0
P-Reactor Basin	577	61 (2)	0	48	1	0
Totals	3,455	14,477 (268)	261	289	91	22

a. Basis for inventory numbers: WSRC-RP-94-110 "SRS Integrated Nuclear Materials and Disposition Plan" revision 0, 1/31/94 (Predecisional Draft).
b. Assemblies include targets and fuel assemblies. Assembly shipments are based on 12 assemblies per shipment.
c. Number of buckets calculated using 54 slugs per bucket. Bucket shipments are based on 3 buckets per shipment.

A.2.9.2.1 Alternative 1 - No Action — The SRS would send the following number of shipments of aluminum-clad fuel sent to the Receiving Basin for Offsite Fuel from:

- K-Reactor Basin - 152;
- L-Reactor Basin - 158;
- P-Reactor Basin - 49;
- Total - 359 shipments.

All nonaluminum-clad fuel would be sent from the Receiving Basin for Offsite Fuel to a reactor basin (a total of 22 shipments).

The number of shipments would be 380. Because fuel handling would occur at both origin and destination, this number would double (i.e., 760 total shipments). Therefore, over 5 years, this alternative would have an average shipping rate of 152 shipments per year.

A.2.9.2.2 Alternative 2 - Decentralization

- **Option 2a - Dry Storage** - For this option, initial shipments would be the same as those for Alternative 1 (760 shipments at a rate of 152 per year). Subsequent shipments from all storage locations to the new dry storage facilities would total 402 shipments. Because fuel handling would occur at both origin and destination, this number would double (i.e., 804 total shipments). Because all fuel would be moved to dry storage within a 5-year period, this total would have an average rate of 161 shipments per year. Adding all shipments would produce a total of 1,564 shipments at a rate of 313 per year.

- **Option 2b - Wet Storage** - For this option, initial shipments would be the same as those for Alternative 1 (760 shipments at a rate of 152 per year). Subsequent shipments from all storage locations to the new wet storage facilities would total 402 shipments for existing SRS fuel. Because the receipt of offsite fuel would continue prior to the relocation of fuel to the new wet storage facilities, an additional 50 shipments would occur [assuming receipt of five shipments per year of offsite fuel (per Volume 1, Appendix I "Offsite Transportation of Spent Nuclear Fuel," 2/15/94 until 2005]. The resulting fuel movement would total 452 shipments. Because fuel handling would occur at both origin and destination, this number would double (i.e., 904 total shipments). Therefore, over 5 years this option would have an average shipping rate of 181 shipments per year. Adding all shipments under this option would produce a total of 1,664 shipments at a rate of 333 per year.

- **Option 2c - Processing** - In this option, all aluminum-clad fuel would move from its present location to the process facilities. All nonaluminum-clad fuel would remain in its present storage locations. The result would be in a total of 380 shipments. As in the previous options, this number would double for a total of 760 shipments. Therefore, over 5 years this option would have an average shipping rate of 152 shipments per year.

A.2.9.2.3 Alternative 3 - Planning Basis

- **Option 3a - Dry Storage** - The movement of materials for this option would be identical to that for Option 2a, resulting in a total of 1,564 shipments at a rate of 313 per year.

- **Option 3b - Wet Storage** - The movement of materials for this option would be identical to that for Option 2b, with the exception of a delay in the receipt of foreign fuel until the new facilities are in operation. This would result in a total of 1,564 shipments at a rate of 313 per year.

- **Option 3c - Processing** - The movement of materials for this option would be identical to that for Option 2c, resulting in a total of 760 shipments at a rate of 152 shipments per year.

A.2.9.2.4 Alternative 4 - Regionalization

- **Option 4a - Dry Storage** - For this option, initial shipments would be the same as Alternative 1 (760 shipments at a rate of 152 per year). Subsequent shipments of the aluminum-clad fuel to the new dry storage facilities would total 380 shipments. (Note: Nonaluminum-clad fuel would be sent from the reactor basins directly off the Site and would not contribute to any further onsite movements.). Because fuel handling would occur at both origin and destination, this number would double (i.e., 760 total shipments). Because all fuel would move to dry storage within about 5 years, this total would have an average shipping rate of 152 shipments per year. Adding all shipments would produce a total of 1,520 shipments at a rate of 304 per year.

- **Option 4b - Wet Storage** - The movement of materials for this option would be identical to that for Option 3b, with the exception of movement of the nonaluminum-clad fuel to the new wet storage facility. This fuel would move off the Site from the reactor basins and would not contribute to any further onsite movements. This would result in a total of 1,520 shipments at a rate of 304 per year.

- **Option 4c - Processing** - The movement of materials for this option would be identical to that for Options 2c and 3c, resulting in a total of 760 shipments at a rate of 152 per year.

- **Option 4d - Dry Storage** - The movement of materials for this option would be identical to those for Options 2a and 3a, resulting in a total of 1,564 shipments at a rate of 313 per year.

- **Option 4e - Wet Storage** - The movement of materials for this option would be identical to that for Option 3b, resulting in a total of 1,564 shipments at a rate of 313 per year.

- **Option 4f - Processing** - The movement of materials for this option would be identical to those for Options 2c, 3c, and 4c, resulting in a total of 760 shipments at a rate of 152 per year.

- **Option 4g - Ship Out** - This option would require the shipping of all spent nuclear fuel at the SRS to a selected regional location. The movement of materials for this option would include the entire spent nuclear fuel inventory at the SRS, resulting in a total of 402 shipments at a rate of 81 per year.

A.2.9.2.5 Alternative 5 - Centralization

- **Option 5a - Dry Storage** - The movement of materials for this option would be identical to those for Options 2a and 3a, resulting in a total of 1,564 shipments at a rate of 313 per year.

- **Option 5b - Wet Storage** - The movement of materials for this option would be identical to that for Option 3b, resulting in a total of 1,564 shipments at a rate of 313 per year.

- **Option 5c - Processing** - The movement of materials for this option would be identical to those for Options 2c, 3c, and 4c, resulting in a total of 760 shipments at a rate of 152 shipments per year.

- **Option 5d - Ship Out** - This option would require the shipping of all spent nuclear fuel at the SRS to a selected central location. The movement of materials for this option would include the entire spent nuclear fuel inventory at the SRS, resulting in a total of 402 shipments at a rate of 81 per year.

A.2.9.3 Methodology for Determination of Offsite Shipping Frequencies. This evaluation determined the total number of offsite shipments using the data contained in Volume 1, Appendix I, "Offsite Transportation of Spent Nuclear Fuel." The total number of Naval Fuel shipments was determined from Table 3 of "Methodology for Adjusting SNF Facility Accident Probabilities and Consequences For Different EIS Alternatives" (dated March 18, 1994).

Naval, foreign, and university shipments would occur throughout the interim management period and could be averaged over the 40-year period covered by this EIS. All other shipments would be averaged over 5 years.

A.2.9.4 Frequency Adjustment Factors for Fuel Handling. For this analysis, DOE assumed the baseline fuel handling rate (events per year) to be the No-Action alternative. For the other alternatives, this evaluation divided the expected spent nuclear fuel handling rate by the baseline spent nuclear fuel handling rate (No Action) to obtain the adjustment factor (see Table A-10).

A.2.9.5 Frequency/Consequence Adjustment Factors Due to Inventory. The No-Action alternative for the SRS would require the storage of 201 MTHM (222 tons) of fuel. Using this amount as the baseline, this evaluation compared the amount of fuel for the other alternatives to the base number, as listed in Table A-11. These adjustment factors can be applied to either a frequency or a consequence, depending on the classification of the accident scenario as listed in Table A-8.

A.3 Chemical Hazard Evaluation

A.3.1 Selection of Reference Chemical Hazard

A review of the same safety analyses used to generate the spectrum of radiological accident scenarios failed to identify a quantitative discussion of chemical hazards. However, each of the safety analyses provided a qualitative discussion of chemical hazards. Thus, Section 5.15.3 discusses chemical hazards associated with existing spent nuclear fuel facilities qualitatively. This qualitative evaluation was determined to be appropriate based on three criteria: sliding scale in proportion to significance, public perception of severity, and long-term effects of chemicals not known. For completeness, a separate risk assessment (WSRC 1993c) provided a quantitative discussion of chemical hazards for the Receiving Basin for Offsite Fuel facility. This assessment described a bounding chemical hazard accident involving the release of nitrogen dioxide vapor.

Table A-10. Fuel handling frequency adjustment factors.

Option Number	Estimated Annual Shipping Rate	Frequency Adjustment Factor
Alternative 1 - No Action		
Option 1	152	Baseline
Alternative 2 - Decentralization		
Option 2a	316	2.08
Option 2b	333	2.19
Option 2c	157	1.03
Alternative 3 - Planning Basis		
Option 3a	375	2.47
Option 3b	375	2.47
Option 3c	216	1.42
Alternative 4 - Regionalization		
Option 4a	421	2.77
Option 4b	421	2.77
Option 4c	269	1.77
Option 4d	394	2.59
Option 4e	394	2.59
Option 4f	234	1.54
Option 4g	160	1.05
Alternative 5 - Centralization		
Option 5a	803	5.28
Option 5b	803	5.28
Option 5c	643	4.23
Option 5d	160	1.05

A.3.2 Hazardous Chemical Inventories

The inventory of hazardous chemicals at each facility was determined by using the "Savannah River Site Tier Two Emergency and Hazardous Chemical Inventory Report" (WSRC 1994a)to get the facility's total chemical inventory, then listing those chemicals that also appeared on the EPA's "List of Lists" (EPA 1990). The chemical inventories listed in Tables A-12 through A-15 represent facilities used for wet storage and/or processing of spent nuclear fuel. The SRS maintains no large-scale dry storage facilities; thus, chemical inventories for dry storage facilities are not listed.

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Centimeter

Inches

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Table A-11. Inventory adjustment factors for each alternative.

Alternative	Inventory[a] (MTHM[b])	Adjustment Factor
No Action	201.44	Baseline
Decentralization	208.60	1.04
Planning Basis	210.51	1.05
Regionalization - A	207.59	1.03
Regionalization - B	263.72	1.31
Centralization	2760.13	13.70

a. (Bulmahn 1993)
b. Metric Tons Heavy Metal; to convert to tons, multiply by 1.1023.

Table A-12. Hazardous chemical inventory for the Receiving Basin for Offsite Fuel.

Chemical	Maximum Daily Amount (Kg)[a]	Average Daily Amount (Kg)
Ethylene glycol	2,981	23
Methyl ethyl ketone	2	2
Nitric acid	4,731	2,365
Phosphoric acid	3,953	3,953
Sodium hydroxide (caustic soda)	5,800	2,900
Sodium nitrite	3,070	1,535

a. To convert kilograms to pounds, multiply by 2.2046.

Table A-13. Hazardous chemical inventory for the reactor basins (typical).

Chemical	Maximum Daily Amount (Kg)[a]	Average Daily Amount (Kg)
Aluminum sulfate (solution)	570	230
Ethylene glycol (thermal arc torch coolant concentrate)	2	2
Hydrogen peroxide	1	1
Nitric acid	75	75
Sodium hydroxide	454	454
Sodium hypochlorite	11	6
Zinc	0.5	0.5

a. To convert kilograms to pounds, multiply by 2.2046.

Table A-14. Hazardous chemical inventory for H-Area.

Chemical	Maximum Daily Amount (Kg)[a]	Average Daily Amount (Kg)
Dichlorodifluoromethane (Freon 12)	227	68
Dichlorodifluoromethane (Racon 12)	227	0
Ethylene glycol	4.0	2.0
Hydrofluoric acid	1	0.5
Hydrogen peroxide	0.5	0.0
Mercury	4,900	4,900
Methyl ethyl ketone	3	3
Nitric acid	10	5
Nitric oxide	1,300	1,300
Phosphorus pentoxide	1	1
Potassium permanganate (Cairox)	200	100
Sodium hydroxide	1	1
Sodium hypochlorite	41	29
Sulfuric acid	1	0.5
Trichlorofluoromethane (Freon 11)	1,150	1,000
Trichlorofluoromethane (Genetron 11)	450	0

a. To convert kilograms to pounds, multiply by 2.2048.

Table A-15. Hazardous chemical inventory for F-Area.

Chemical	Maximum Daily Amount (Kg)[a]	Average Daily Amount (Kg)
Dichlorodifluoromethane (Freon 12)	1	0.5
Dichlorodifluoromethane (Racon 12)	1	0
Ethylene glycol	4	2
Hydrofluoric acid	1,177	1,177
Potassium permanganate	3	1
Sodium hydroxide	0.5	—
Sodium hypochlorite	7	4
Sulfuric acid	30	—
Trichlorofluoromethane (Freon 11)	900	450

a. To convert kilograms to pounds, multiply by 2.2048.

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