COVER SHEET

RESPONSIBLE AGENCY: U.S. Department of Energy (DOE)

TITLE: Savannah River Site, Spent Nuclear Fuel Management Final Environmental Impact Statement (DOE/EIS-0279) EC

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ABSTRACT: The proposed DOE action considered in this environmental impact statement (EIS) is to implement appropriate processes for the safe and efficient management of spent nuclear fuel and targets at the Savannah River Site (SRS) in Aiken County, South Carolina, including placing these materials in forms suitable for ultimate disposition. Options to treat, package, and store this material are discussed. The material included in this EIS consists of approximately 68 metric tons heavy metal (MTHM) of spent nuclear fuel (20 MTHM of aluminum-based spent nuclear fuel at SRS, as much as 28 MTHM of aluminum-clad spent nuclear fuel from foreign and domestic research reactors to be shipped to SRS through 2035, and 20 MTHM of stainless-steel or zirconium-clad spent nuclear fuel and some Americium/Curium Targets stored at SRS. EC

Alternatives considered in this EIS encompass a range of new packaging, new processing, and conventional processing technologies, as well as the No Action Alternative. A preferred alternative is identified in which DOE would prepare about 97 percent by volume (about 60 percent by mass) of the aluminum-based fuel for disposition using a melt and dilute treatment process. The remaining 3 percent by volume (about 40 percent by mass) would be managed using chemical separation. Impacts are assessed primarily in the areas of water resources, air resources, public and worker health, waste management, socioeconomic, and cumulative impacts.

PUBLIC INVOLVEMENT: DOE issued the Draft Spent Nuclear Fuel Management EIS on December 24, 1998, and held a formal public comment period on the EIS through February 8, 1999. In preparing the Final EIS, DOE considered comments received via mail, fax, electronic mail, and transcribed comments made at public hearings held in Columbia, S.C. on January 28, 1999, and North Augusta, S.C. on February 2, 1999. Completion of the Final EIS has been delayed because DOE has performed additional analyses of the melt and dilute technology, discussed in Chapter 2 and Appendix G. Comments received and DOE's responses to those comments are found in Appendix G of the EIS. EC

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ACRONYMS, ABBREVIATIONS, AND USE OF SCIENTIFIC NOTATION

Acronyms

AAQS	ambient air quality standard
ALARA	as low as reasonably achievable
CFR	Code of Federal Regulations
CO	carbon monoxide
D&D	decontamination and decommissioning
DNFSB	Defense Nuclear Facilities Safety Board
DOE	U.S. Department of Energy
DWPF	Defense Waste Processing Facility
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
ES&H	environment, safety and health
FR	Federal Register
GMODS	glass material oxidation and dissolution system
HEPA	high-efficiency particulate air [filter]
HEU	highly enriched uranium
HLW	high-level waste
IMNM	Interim Management of Nuclear Material
INEEL	Idaho National Engineering and Environmental Laboratory
ISO	International Organization for Standardization
LCF	latent cancer fatality
LEU	low enriched uranium
MCL	maximum contaminant level
MEI	maximally exposed (offsite) individual
MTHM	metric tons of heavy metal

NAS	National Academy of Sciences
NCRP	National Council on Radiation Protection and Measurements
NESHAP	national emission standards for hazardous air pollutants
NIMS	nuclear incident monitoring system
NO_X	nitrogen oxides
NPDES	national pollutant discharge elimination system
NRC	Nuclear Regulatory Commission
O_3	ozone
OSHA	Occupational Safety and Health Administration
PM_{10}	particulate matter less than 10 microns in diameter
RBOF	Receiving Basin for Offsite Fuel
RINM	reactor irradiated nuclear materials
ROD	Record of Decision
SCDHEC	South Carolina Department of Health and Environmental Control
SMDF	Saltstone Manufacturing and Disposal Facility
SNF	spent nuclear fuel
SO_2	sulfur dioxide
SRI	Savannah River Natural Resources Management and Research Institute
SRS	Savannah River Site
TRIGA	Training Research Isotope general atomic [spent fuel]
TSP	total suspended particulates
TSS	total suspended solids
VLEU	very low enriched uranium
WSRC	Westinghouse Savannah River Company

Abbreviations for Measurements

cfm	cubic feet per minute
cfs	cubic feet per second = 448.8 gallons per minute = 0.02832 cubic meter per second
cm	centimeter
gpm	gallons per minute
kg	kilogram
L	liter = 0.2642 gallon
lb	pound = 0.4536 kilogram
mg	milligram
µCi	microcurie
µg	microgram
pCi	picocurie
°C	degrees Celsius = 5/9 (degrees Fahrenheit – 32)
°F	degrees Fahrenheit = 32 + 9/5 (degrees Celsius)

Use of Scientific Notation

Very small and very large numbers are sometimes written using "scientific notation" or "E-notation" rather than as decimals or fractions. Both types of notation use exponents to indicate the power of 10 as a multiplier (i.e., 10^n, or the number 10 multiplied by itself "n" times; 10^{-n}, or the reciprocal of the number 10 multiplied by itself "n" times).

For example: $10^3 = 10 \times 10 \times 10 = 1{,}000$

$$10^{-3} = \frac{1}{10 \times 10 \times 10} = 0.001$$

In scientific notation, large numbers are written as a decimal between 1 and 10 multiplied by the appropriate power of 10:

4,900 is written $4.9 \times 10^3 = 4.9 \times 10 \times 10 \times 10 = 4.9 \times 1{,}000 = 4{,}900$
0.049 is written 4.9×10^{-2}
1,490,000 or 1.49 million is written 1.49×10^6

A positive exponent indicates a number larger than or equal to one, a negative exponent indicates number less than one.

In some cases, a slightly different notation ("E-notation") is used, where "$\times$ 10" is replaced by "E" and the exponent is not superscripted. Using the above examples

$4{,}900 = 4.9 \times 10^3 = 4.9\text{E+}03$
$0.049 = 4.9 \times 10^{-2} = 4.9\text{E-}02$
$1{,}490{,}000 = 1.49 \times 10^6 = 1.49\text{E+}06$

Metric Conversion Chart

To convert into metric			To convert out of metric		
If you know	**Multiply by**	**To get**	**If you know**	**Multiply by**	**To get**
Length					
inches	2.54	centimeters	centimeters	0.3937	inches
feet	30.48	centimeters	centimeters	0.0328	feet
feet	0.3048	meters	meters	3.281	feet
yards	0.9144	meters	meters	1.0936	yards
miles	1.60934	kilometers	kilometers	0.6214	miles
Area					
sq. inches	6.4516	Sq. centimeters	sq. centimeters	0.155	sq. inches
sq. feet	0.092903	sq. meters	sq. meters	10.7639	sq. feet
sq. yards	0.8361	sq. meters	sq. meters	1.196	sq. yards
acres	0.0040469	sq. kilometers	sq. kilometers	247.1	acres
sq. miles	2.58999	sq. kilometers	sq. kilometers	0.3861	sq. miles
Volume					
fluid ounces	29.574	milliliters	milliliters	0.0338	fluid ounces
gallons	3.7854	liters	liters	0.26417	gallons
cubic feet	0.028317	cubic meters	cubic meters	35.315	cubic feet
cubic yards	0.76455	cubic meters	cubic meters	1.308	cubic yards
Weight					
ounces	28.3495	grams	grams	0.03527	ounces
pounds	0.4536	kilograms	kilograms	2.2046	pounds
short tons	0.90718	metric tons	metric tons	1.1023	short tons
Temperature					
Fahrenheit	Subtract 32 then multiply by 5/9ths	Celsius	celsius	Multiply by 9/5ths, then add 32	Fahrenheit

Metric Prefixes

Prefix	Symbol	Multiplication Factor
exa-	E	1 000 000 000 000 000 000 = 10^{18}
peta-	P	1 000 000 000 000 000 = 10^{15}
tera-	T	1 000 000 000 000 = 10^{12}
giga-	G	1 000 000 000 = 10^{9}
mega-	M	1 000 000 = 10^{6}
kilo-	k	1 000 = 10^{3}
centi-	c	0.01 = 10^{-2}
milli	m	0.001 = 10^{-3}
micro-	µ	0.000 001 = 10^{-6}
nano-	n	0.000 000 001 = 10^{-9}
pico-	p	0.000 000 000 001 = 10^{-12}
femto-	f	0.000 000 000 000 001 = 10^{-15}
atto-	a	0.000 000 000 000 000 001 = 10^{-18}

FOREWORD

The U.S. Department of Energy (DOE) published a Notice of Intent (NOI) to prepare this environmental impact statement (EIS) on December 31, 1996 (61 FR 69085). As described in the NOI, DOE's proposal in general terms is to implement appropriate actions to manage safely and efficiently spent nuclear fuel (SNF) and targets that are currently located or expected to be received at the Savannah River Site (SRS), including placing these materials in forms suitable for disposition. This EIS assesses the potential environmental impacts associated with storing, treating, and packaging these materials, including onsite transportation activities.

The NOI requested public comments and suggestions for DOE to consider in its determination of the scope of the EIS, and announced a public scoping period that ended on March 3, 1997. DOE held a scoping meeting in North Augusta, South Carolina on January 30, 1997. During the scoping period, individuals, organizations, and government agencies submitted 118 comments that DOE considered applicable to the management of SNF at the SRS.

Transcripts of public testimony, copies of scoping letters, scoping comments and DOE responses to those comments, and reference materials cited in the EIS are available for review in the DOE Public Reading Room, University of South Carolina at Aiken, Gregg-Graniteville Library, University Parkway, Aiken, South Carolina.

A Notice of Availability for the Draft EIS appeared in the *Federal Register* on December 24, 1998. Public meetings to discuss and receive comments on the Draft EIS were held on Thursday, January 28, 1999 in Columbia, S.C. and on Tuesday, February 2, 1999 in North Augusta, S.C. The public comment period ended on February 8, 1999. Comments and DOE responses to comments are in Appendix G.

Changes from the Draft EIS are indicated in this Final EIS by vertical change bars in the margin. In cases where changes were made in response to comments, the comment number (as listed in Appendix G) is listed next to the vertical change bar. Many of the technical changes are the result of the availability of updated information since publication of the Draft EIS.

DOE has prepared this EIS in accordance with the National Environmental Policy Act (NEPA) regulations of the Council on Environmental Quality (40 CFR 1500-1508) and DOE NEPA Implementing Procedures (10 CFR 1021). This EIS identifies the methods used for analyses and the scientific and other sources of information consulted. In addition, it incorporates, directly or by reference, available results of ongoing studies. The organization of the EIS is as follows:

- Chapter 1 describes the purpose and need for SNF management at the SRS (i.e., to develop and implement a safe and efficient management strategy that includes preparing SNF for ultimate disposition), and describes the types of SNF to which the EIS applies.

- Chapter 2 identifies the alternatives that DOE is considering for management of SNF at the SRS.

- Chapter 3 describes the SRS environment as it relates to the alternatives described in Chapter 2.

- Chapter 4 assesses the potential environmental impacts of the alternatives for construction activities, normal operations, and accidents.

- Chapter 5 discusses the cumulative impacts of SNF management actions in relation to

impacts of other past, present, and foreseeable future activities at the SRS.

- Chapter 6 identifies irreversible or irretrievable resource commitments.

- Chapter 7 discusses regulatory requirements, including applicable statutes, DOE Orders, and state and Federal regulations.

- Appendix A describes the technologies that DOE considered for implementing the SNF management alternatives described in Chapter 2.

- Appendix B describes previously identified facility vulnerabilities specific to SRS SNF management, their recommended corrective actions, and the current status of those corrective actions.

- Appendix C describes the SNF assigned to SRS for management and the categories into which DOE has grouped these fuels.

- Appendix D provides detailed descriptions of accidents that could occur at SRS facilities during the management of SNF.

- Appendix E describes assumed durations for each SNF management activity necessary to implement the alternatives described in Chapter 2.

- Appendix F lists estimated incremental nonradiological air concentrations attributable to SNF management activities.

- Appendix G describes public comments received on the Draft EIS and DOE responses. TC

Change Bars

Changes from the Draft EIS are indicated in this Final EIS by vertical change bars in the margin. The bars are marked TC for technical changes, EC for editorial changes, or if the change was made in response to a public comment, the designated comment number is as listed in Appendix G of the EIS. EC

CHAPTER 1. INTRODUCTION

The management of spent nuclear fuel (SNF) has been an integral part of the mission of the Savannah River Site (SRS) for more than 40 years. Until the early 1990s, SNF management consisted primarily of short-term onsite storage and processing in the SRS chemical separation facilities to produce strategic nuclear materials. EC

With the end of the Cold War, the U.S. Department of Energy (DOE) decided to phase out processing of SNF for the production of nuclear weapons materials (DOE 1992). Therefore, the management strategy for this fuel has shifted from short-term storage and processing for the recovery of highly-enriched uranium and transuranic isotopes to stabilization, when necessary, and storage pending final disposition that includes preparing aluminum-based SNF for placement in any potential geologic repository. In addition to the fuel already onsite, the SRS will receive SNF from foreign research reactors until 2009 and from domestic research reactors until, potentially, 2035. As a result, the safe and efficient management of SNF will continue to be an important SRS mission. EC TC

This EIS evaluates the potential environmental impacts of DOE's proposed plans for managing SNF assigned to SRS.

1.1 Background

1.1.1 HISTORIC MISSIONS

The U.S. Atomic Energy Commission, a DOE predecessor agency, established the SRS in the early 1950s. The Site occupies an area of approximately 300 square miles (800 square kilometers) adjacent to the Savannah River, primarily in Aiken and Barnwell Counties in South Carolina. It is approximately 25 miles (40 kilometers) southeast of Augusta, Georgia, and 20 miles (32 kilometers) south of Aiken, South Carolina (Figure 1-1).

For the past 40 years the SRS mission has been the production of special radioactive isotopes to support national programs. Historically, the primary Site mission was the production of strategic isotopes (plutonium-239 and tritium) for use in the development and production of nuclear weapons. The SRS produced other isotopes (e.g., californium-252, plutonium-238, americium-241) to support research in nuclear medicine, space exploration, and commercial applications. DOE produced these isotopes in the five SRS production reactors. After the material was produced at the SRS, it was shipped to other DOE sites for fabrication into desired forms.

1.1.2 FUEL CYCLE

The material in the SRS reactors consisted of nuclear fuel and targets. The nuclear fuel was enriched uranium that was alloyed with aluminum and then clad with aluminum. The targets were either oxides or metallic forms of various isotopes such as neptunium-237 or uranium-238 that were clad with aluminum. Fuel and targets were fabricated at the SRS and placed in the reactors, and then the reactors operated to create the neutrons necessary to transmute the target material. For example, neptunium-237 targets were irradiated to produce plutonium-238, a material used by the National Aeronautics and Space Administration as a power source for deep space probes. After irradiation, the fuel and targets (collectively referred to as spent nuclear fuel) were removed from the reactors and placed in water-filled basins for short-term storage, about 12 to 18 months, before they were processed in the SRS separations facilities. Figure 1-2 shows the historic fuel and target cycle. EC EC

During processing, SNF was chemically dissolved in F or H Canyon to recover the uranium and transuranic isotopes. The recovered material was used in nuclear weapons programs or EC

Figure 1-1. Location of the Savannah River Site.

Figure 1-2. Historic nuclear materials production cycle at the Savannah River Site.

for commercial applications. The remaining residue from the fuel, high-level radioactive waste consisting primarily of fission products and cladding in liquid form, was transferred to large steel tanks for storage. The high-level waste is currently being vitrified in the Defense Waste Processing Facility at the SRS to prepare it for disposal in any potential geologic repository. TC

1.1.3 CHANGING MISSIONS

With the end of the Cold War there was a decreased need for the strategic nuclear material that was produced at the SRS. In 1992, the Secretary of Energy directed that processing operations be phased out throughout the DOE complex, effectively halting the SRS mission to produce strategic nuclear materials such as plutonium-239. However, SNF and targets from previous production reactor irradiation cycles remained in storage at K-, L-, C-, and P-Reactor Disassembly Basins. (Chapter 2 describes SRS SNF storage facilities.)

In addition to nuclear material production missions, another mission for the SRS was (and continues to be) the receipt of SNF from DOE, domestic, and foreign research reactors. These reactors were operated by DOE, universities, and research institutions for educational and research purposes and to produce isotopes for nuclear medicine. Historically, SNF from these reactors was stored in the Receiving Basin for Offsite Fuel at SRS. In the past, much of the research reactor SNF was processed in the same manner as spent fuel from SRS production reactors. However, with the end of the Site's strategic nuclear materials production mission, SNF from research reactors has been accumulating in the Receiving Basin for Offsite Fuel and in the L-Reactor Disassembly Basin. EC EC

Some of the research reactor spent nuclear fuel sent to SRS was not aluminum based. Because DOE did not have the capability to process that type of SNF at SRS, it was placed in wet storage at the Receiving Basin for Offsite Fuel, where it remains in storage. EC

By 1995 DOE was storing about 195 metric tons heavy metal (MTHM [metric tons heavy metal] – the mass of uranium in the fuel or targets, excluding cladding, alloy materials, and structural materials) – of aluminum-based SNF in the SRS reactor disassembly basins and the Receiving Basin for Offsite Fuel. DOE also was storing about 20 MTHM of non-aluminum-based SNF in the Receiving Basin for Offsite Fuel. EC

1.1.4 STABILIZATION

DOE has taken action to stabilize about 175 MTHM of the 195 MTHM of aluminum-based SNF that was in storage at SRS in 1995. DOE decided to stabilize this material following completion of the *Interim Management of Nuclear Materials Environmental Impact Statement* (DOE 1995a). The primary purpose of the actions described in that environmental impact statement (EIS) was to correct or eliminate potential health and safety vulnerabilities related to some of the methods used to store nuclear materials (including SNF) at SRS. The vulnerable SNF had been stored in wet storage basins with poor water quality. The poor water quality resulted in corrosion and failure of the cladding on the fuel and subsequent releases of radioactive fission products to the water of the storage basins. In 1996, SRS began stabilizing vulnerable aluminum-based uranium metal SNF in F Canyon. That work is complete. Vulnerable aluminum-based SNF still is being stabilized in H Canyon and that work is expected to continue through 2002. In the *Interim Management of Nuclear Materials EIS* (DOE 1995a), DOE identified 20 MTHM (out of 195 MTHM) of aluminum-based SNF at SRS that was "stable," i.e., that likely could be safely stored for about 10 more years, pending decisions on final disposition. That 20 MTHM of aluminum-based SNF is included in this EIS. EC

1.1.5 SPENT NUCLEAR FUEL CONSOLIDATION

In May 1995, DOE decided (60 FR 28680) under the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental*

Restoration and Waste Management Programs Final Environmental Impact Statement to consolidate existing and newly generated SNF at three existing Departmental sites based on the fuel type, pending future decisions on ultimate disposition. Specifically, DOE decided that existing Hanford production reactor fuel would remain at Hanford, aluminum-based SNF (excluding the aluminum-based SNF at Hanford) would be consolidated at SRS, and non-aluminum-based SNF would be consolidated at the Idaho National Engineering and Environmental Laboratory (INEEL). DOE stated that decisions on preparing the SNF for final disposition would be made under site-specific National Environmental Policy Act evaluations. As a result of DOE's decision to consolidate SNF storage, DOE will transfer 20 MTHM of non-aluminum-based SNF from SRS to INEEL and will transfer about 5 MTHM of aluminum-based SNF at INEEL to SRS. DOE estimates these transfers could begin about 2009 and may be completed by 2017. Thus, the non-aluminum-based SNF at SRS and the aluminum-based SNF from INEEL that will be transferred to the SRS are included in this EIS. Additionally, as a result of the consolidation decision DOE reached under the *Programmatic Spent Nuclear Fuel Management and Idaho mental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995b), SRS could receive about 5 MTHM of aluminum-based SNF from domestic research reactors. Shipments from domestic research reactors could continue through 2035. Material expected to be received from domestic research reactors is included in this EIS.

In May 1996, DOE announced a decision (61 FR 25092) under the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (Nonproliferation Policy and Spent Fuel EIS) to accept about 18 MTHM of aluminum-based SNF containing uranium of United States origin from foreign research reactors for management in the United States at the SRS. The receipt of foreign research reactor SNF at SRS is now underway and receipts are scheduled to be completed by 2009. The 18 MTHM of foreign research reactor SNF that could be received at SRS is included in the scope of this EIS. (Recent decisions by some foreign research reactor operators have reduced the quantity of SNF expected to be shipped to SRS from about 18 MTHM to about 14 MTHM; however, the 18 MTHM projection is used for analysis purposes in this EIS because foreign research reactor operators still have the option to ship to the United States.)

1.1.6 PREPARATION FOR DISPOSITION

In summary, the total quantity of aluminum-based SNF at SRS that must be managed and prepared for disposition is as follows: 20 MTHM in existing SRS wet storage basins; about 10 MTHM to be received from INEEL and domestic research reactors; and about 18 MTHM to be received from foreign research reactors. Additionally, SRS must manage about 20 MTHM of non-aluminum-based SNF until it is transferred to INEEL.

1.2 Purpose and Need for Action

DOE anticipates placing most of its aluminum-based SNF inventory in a geologic repository after treatment or repackaging. However, DOE does not expect any geologic repository to be available until at least 2010 and shipments from DOE sites would not begin until about 2015. Until a repository is available, the Department intends to develop and implement a safe and efficient SNF management strategy that includes preparing aluminum-based SNF stored at SRS or expected to be shipped to SRS for disposition offsite. DOE is committed to avoiding indefinite storage at the SRS of this nuclear fuel in a form that is unsuitable for final disposition. Therefore, DOE needs to identify management technologies and facilities for storing and treating this SNF in preparation for final disposition.

1.3 Scope

This EIS evaluates potential environmental impacts from managing SNF that currently is lo-

cated or expected to be located at SRS. The evaluation includes impacts from the construction and operation of facilities (either new or modified existing facilities) that would be used to receive, store, treat, and package SNF in preparation for ultimate disposition. Onsite transportation impacts are considered, however, no impacts associated with transporting SNF to SRS are included, because these impacts have been covered in other EISs. The potential impacts of transporting SNF to a geologic repository are discussed (in Chapter 4) for completeness but no decisions related to transporting SNF offsite will be made under this EIS. Transportation of SNF (and high-level waste) to a federal repository will be addressed in the EIS for a federal repository (see Section 1.6). The Yucca Mountain EIS is being prepared as part of the process to determine whether to recommend the Yucca Mountain site as the site of the Nation's first geologic repository for SNF and high-level radioactive waste.

In this EIS, DOE is evaluating the management of about 48 MTHM of aluminum-based SNF for treatment and storage (20 MTHM of aluminum-based SNF stored at SRS and about 28 MTHM of aluminum-based SNF from foreign and domestic research reactors that could be shipped to SRS until 2009 and from domestic research reactors that could be shipped to SRS until 2035).

DOE also evaluates transferring 20 MTHM of non-aluminum-clad spent nuclear fuel currently stored in the Receiving Basin for Offsite Fuel at SRS to a new dry storage facility at SRS. This transfer would occur only if a dry storage facility were built as part of the implementation of a new treatment technology to prepare aluminum-based spent nuclear fuel for disposition (potential technologies are discussed in Section 2.2) and if the dry storage facility became operational before the non-aluminum-clad fuel was transferred to the INEEL. The transfer to dry storage would occur after the fuel had been relocated from the Receiving Basin for Offsite Fuel to the L-Reactor Disassembly Basin in support of activities necessary to phase out the use of the Receiving Basin for Offsite Fuel by fiscal year 2007.

This EIS does not evaluate the impacts of managing the non-aluminum-clad fuel at INEEL or of transporting the fuel to INEEL. These impacts were documented in the SNF programmatic EIS (PEIS) (DOE 1995b) and were evaluated as part of the process DOE used to decide to consolidate the storage of non aluminum-clad spent nuclear fuel at the INEEL.

SRS is storing Mark-51 and other targets in the Receiving Basin for Offsite Fuel (RBOF) in the Site's H-Area. This EIS evaluates the impacts of continuing to store the Mark-51 and other targets in RBOF, and evaluates an alternative of transferring them to dry storage to provide flexibility in material management operations.

DOE is evaluating potential uses for this material and the operations and facilities that would be necessary. The Mark-51 and other targets (described in Section 1.5 of this EIS) contain americium and curium isotopes that could be used to produce elements with higher atomic numbers such as californium-252. Californium-252 is used as a neutron source for radiography and in the treatment of certain types of cancer and for research in basic chemistry, nuclear physics, and solid-state chemistry. If DOE were to determine that a programmatic need for this material exists, the targets would continue to be stored at the SRS pending preparations to ship them to another DOE facility where isotope production capability currently exists or could be constructed and operated. SRS does not have isotope production capability.

This EIS does not evaluate the impacts of utilizing target material for programmatic purposes such as production of californium. DOE would perform the appropriate National Environmental Policy Act review to evaluate the impacts of shipment of the targets to an isotope production facility and of construction (or modification) and operation of the production facility, should such a programmatic purpose be identified.

DOE is storing the Mark-18 targets in wet basins at the SRS. These targets are similar to the Mark-51 and other targets in that they contain

americium and curium that could be used to produce elements with higher atomic numbers such as californium-252. They are different from the small (about two feet in length) Mark-51 and other targets because the Mark 18s are about 12 feet long and therefore have different requirements for storage, transportation and use. As is the case with the Mark-51 and other targets, DOE is not proposing any actions that would lead to programmatic use of the Mark-18 targets at this time. Because of their length, the Mark-18 targets would have to be reduced in size for use in production facilities at another DOE facility or transfer to dry storage at the SRS. This EIS considers only continued wet storage of Mark-18 targets. However, the Interim Management of Nuclear Materials EIS (which is incorporated herein by reference) considered the alternative of processing the Mark-18 targets in the SRS canyons, should they present potential health and safety vulnerabilities. See Section 1.5 of this EIS for more information.

1.4 Decisions to be Based on this EIS

DOE expects to make the following decisions on the management and preparation of SNF for storage and ultimate disposition.

- The selection of the appropriate treatment or packaging technologies to prepare aluminum-based SNF that is to be managed at SRS.
- Whether DOE should construct new facilities or use existing facilities to store and treat, or package aluminum-based SNF that is expected to be managed at SRS.
- Whether DOE should repackage and dry-store stainless-steel and zirconium-clad SNF pending shipment to the Idaho National Engineering and Environmental Laboratory.
- Whether DOE should repackage and dry-store Mark-51s and other americium/curium targets in the event dry-storage capability becomes available at SRS.

1.5 Spent Nuclear Fuel Groups

This section introduces the basic terminology for describing SNF and provides more information on the approximately 68 MTHM of SNF subject to analysis in this EIS.

DOE has categorized the spent fuel considered in this EIS into six groups (Group A through Group F). The categorization is based on such characteristics as fuel size, physical or chemical properties, or radionuclide inventories. DOE grouped the fuel to distinguish how it could apply the management alternatives evaluated in the EIS (Section 2.2). Table 1-1 lists the fuel groups and the amount of fuel in each group. Appendix C provides more detailed information regarding fuel types, quantities, locations, radionuclide inventories, and curie content.

The aluminum-based fuels currently stored at SRS include some fuels that were not originally aluminum-clad (EBR-II and Sodium Breeder Experimental Reactor Fuel). Additionally, the aluminum-based category consists of one element not yet received but due to be shipped to SRS (the Advanced Reactivity Measurement Facility Core Filter Block). Most of the fuels that were not originally aluminum-clad (but are included under this EIS's major category of aluminum-based fuel) have been declad and placed in aluminum cans. In their present form they can be processed at the SRS through the existing technologies on site. Other fuels at SRS which are non-aluminum-clad fuels cannot be processed in their existing form using the existing technologies and are characterized in this EIS as non-aluminum-based fuel. The Core Filter Block is included under the category of

Table 1-1. Spent nuclear fuel groups.

Fuel group	Volume (MTRE)[a]	Mass (MTHM)[b]
A. Uranium and Thorium Metal Fuels	610	19
B. Material Test Reactor-Like Fuels	30,800	20
C. HEU/LEU[c] Oxides and Silicides Requiring Resizing or Special Packaging	470[d]	8
D. Loose Uranium Oxide in Cans	NA	0.7
E. Higher Actinide Targets	NA	<0.1
F. Non-Aluminum-Clad Fuels[e]	1,900	20.4
Total	33,780	68.2

NA = Not applicable

a. MTRE = Materials test reactor equivalent. An MTRE is a qualitative estimate of SNF volume that provides information on the amount of space needed for storage. An MTRE of Materials Test Reactor-Like Fuels would usually be one fuel assembly measuring about 3 inches by 3 inches by 2 feet long.

b. MTHM = Metric tons of heavy metal.

c. HEU = highly enriched uranium; LEU = low enriched uranium.

d. Fuel group also includes about 2,800 pins, pin bundles, and pin assemblies.

e. This fuel group will be shipped to Idaho National Engineering and Environmental Laboratory. It will not be treated at SRS.

aluminum-based fuel since the most practical way of dealing with it (based on its unique configuration) is to process it utilizing the existing technology at SRS.

Uranium and Thorium Metal Fuels (Group A):

This group consists of fuels from the Experimental Breeder Reactor-II and the Sodium Reactor Experiment, as well as a core filter block from the Advanced Reactivity Measurement Facility at INEEL (that is scheduled to be transferred to SRS). This group also includes unirradiated Mark-42 targets that were manufactured from plutonium oxide-aluminum powder metal and formed into tubes that were clad with aluminum

The Experimental Breeder Reactor-II fuel and Sodium Reactor Experiment fuel are uranium metal that has been declad and stored in canisters in the Receiving Basin for Offsite Fuel. The declad fuel presents a potential health and safety vulnerability. These fuels have cores of reactive metals that were exposed when the fuel cladding was removed. Any contact of the reactive metal core with water would lead to relatively rapid oxidation of the core and disintegration of the fuel. Should the existing storage containers leak, the metal fuel would corrode and release fission products to the water of the storage basin. Once the metal of the fuel is wetted, simply repackaging the fuel in a watertight container would not arrest the corrosion and, in fact, could exacerbate storage concerns since potentially explosive hydrogen gas would continue to be generated inside the storage canister as the fuel continued to corrode. Water intrusion and subsequent fuel corrosion has already occurred with one Experimental Breeder Reactor-II canister stored in the Receiving Basin for Offsite Fuel. That material was processed in F Canyon to eliminate the problem. In the event that leaks were detected in any additional canisters prior to processing/treatment in accordance with decisions reached under this EIS, DOE would process those canisters in an SRS canyon facility. This management approach is consistent with the Records of Decision reached under the *Interim Management of Nuclear Materials Final Environmental Impact Statement* for other uranium metal SNF stored in the Receiving Basin for Offsite Fuel at the SRS. The *Interim Management of Nuclear Materials EIS* deferred decisions on the materials that did not pose

immediate health and safety vulnerabilities because they were considered to be stable for 10 years and DOE wanted to provide the public an opportunity to comment as part of the overall planning for SNF at SRS.

The unirradiated Mark-42 targets were manufactured from plutonium oxide-aluminum powder metal and formed into tubes that were clad with aluminum. The plutonium oxide and aluminum were pressed together in the manufacturing process. As a result, the unirradiated targets are less durable than uranium-aluminum alloy SNF because of the particulate nature of the plutonium oxide but more durable (i.e., less reactive) than uranium metal SNF since the plutonium is already in oxide form. The unirradiated Mark-42 targets present a potential safety and health vulnerability in that should the cladding of these targets be breached, the plutonium oxide could migrate to the water of the storage basin.

The core filter block at INEEL is made of depleted uranium and was used as a neutron "filter" for reactivity experiments. As a result, the filter was subject to relatively short (or low-power level) exposure times in the test reactor and is only slightly irradiated. The core filter block contains cylindrical sleeves of various corrosion resistant metals at different diameters within the filter block.

EC

DOE is unaware of any health or safety concerns related to the core filter block. The core filter block is a unique assembly in that it includes materials that would not be compatible with the melt and dilute process for aluminum-based SNF. Additionally, the core filter block is composed mainly of depleted uranium and has been exposed to relatively low power so it contains very little fissile material or fission products. Processing would not extend the time for planned canyon operations, would not generate recovered fissile material, and would produce only a few kilograms of depleted uranium.

L2-14

EC

There is uncertainty regarding the acceptability of the material in this fuel group in its current form into a repository due to the reactive nature of uranium metal or the particulate nature of some of the material. The oxidation or burning of the metal in the repository could cause damage and spread radioactive particles throughout the repository. Although somewhat less reactive than pure metals, the uranium and thorium metal fuels discussed in this EIS (Group A) would need special attention to mitigate their reactivity.

EC

This group accounts for approximately 2.0 percent of the volume of aluminum-based fuel that DOE is likely to manage at the SRS from now until 2035. Because the fuel in Group A is made of unalloyed metal (i.e., it contains little or no aluminum), it is more dense than most of the other spent fuel considered in this EIS. As a result, this small volume of fuel contains about 40 percent of the mass of heavy metal.

Materials Test Reactor-Like Fuels (Group B):

This group consists primarily of Materials Test Reactor fuels and other fuels of similar size and composition. Most research reactors – foreign and domestic – use Materials Test Reactor fuel, which has a flat or curved plate design. Figure 1-3 shows a typical Materials Test Reactor fuel assembly. Although these fuels come in a variety of shapes and compositions, the active fuel region is typically about 2 feet (0.6 meter) long and the overall assembly is about 4 feet (1.2 meters) long. The cross-section of an assembly is approximately square, about 3 inches (8 centimeters) on a side.

These fuels vary in enrichment. Approximately 70 percent of the Group B assemblies are highly enriched uranium, and the remainder are low enriched uranium. They are uranium-aluminum, uranium oxide-aluminum, or uranium silicide-aluminum alloy; all types are clad with aluminum. Group B accounts for approximately 97 percent of the volume of aluminum-based SNF that DOE will manage at SRS between now and 2035. DOE considers that there are no currently known health and safety vulnerabilities for this material that would preclude wet storage pending the operation of a new treatment technology.

TC

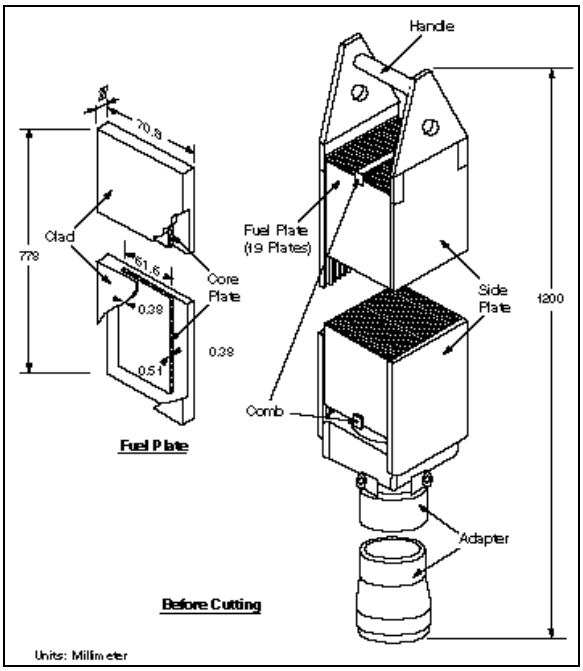

Figure 1-3. Typical Materials Test Reactor fuel assembly.

Although some Group B fuels are stored at SRS in the Receiving Basin for Offsite Fuel or in L Disassembly Basin, at present most are at domestic universities, foreign research reactors, and DOE research facilities pending shipment to the Site. All of the Group B fuels that are currently stored at SRS are "intact." The good condition of the cladding and the durability of the alloyed fuel at SRS provide a high degree of confidence that the fuel will not degrade during storage and that actions to correct potential health and safety vulnerabilities will not be necessary before treatment using the technology that DOE proposes to select under the record of decision from this EIS. DOE expects this will be true for most of the foreign and domestic research reactor SNF included in Group B that is yet to be shipped to SRS. However, if DOE determines that any of the Group B fuel presents a health and safety vulnerability, DOE would evaluate the situation and take appropriate action that could include canning the problem fuel or processing the fuel in one of the SRS canyon facilities. This management approach is consistent with the Record of Decision reached under the *Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel.*

HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (Group C):

Fuels in this group are similar in composition to Group B fuels in that they are aluminum-based, highly enriched uranium (HEU) and low enriched uranium (LEU) oxides and silicides, but their size or shape might preclude packaging them in the disposal canisters proposed for use in a repository without resizing or special packaging considerations. Some fuel in this group is smaller in diameter and longer than Group B fuels or is larger than Group B fuels in both diameter and length; it often comes in odd shapes such as a 1.5-foot by 3-foot (0.46-meter by 0.9-meter) cylinder or a sphere with a diameter of 29 inches (74 centimeters). DOE would have to disassemble or use other volume-reduction activities to place such fuels in a nominal 17-inch direct co-disposal canister (see Section 2.2). At present, much of this fuel is at other DOE sites and in other countries but is scheduled to be received at SRS. EC EC

DOE expects that most of the fuel in this category is intact and would be managed as described above for Group B fuels. However, a small amount is not intact. That material consists of some fuel and one target that were cut or sectioned for research purposes. After the research was completed, the fuel and target pieces were canned in 14 cans and placed in wet storage. The origin and location of this material is discussed in Appendix C, Table C-3. The sectioned fuel and target present a potential health and safety vulnerability similar to that of the Group A fuel discussed previously. If a storage can were to leak, DOE would address the problem as described for the Group A fuel to prevent the release of fission products and particulate material to the water of a storage basin. Additionally, the current form of the fuel (i.e., failed) may not be acceptable in a repository because its integrity has been compromised. EC

Together Group B and Group C fuels represent 97 percent of all fuel to be managed at SRS, and 93 percent of the total fuel at SRS (including Group F fuels which will be shipped to Idaho National Engineering and Environmental Laboratory without treatment at SRS).

Loose Uranium Oxide in Cans (Group D):

This group consists of loose uranium oxide with fission products distributed through the material that has been stored in aluminum cans. This material, in its current particulate form, probably would not be acceptable for disposal in a repository because it is not in a tightly bound metal or ceramic matrix. Therefore, this group probably would require special packaging and/or treatment. Group D fuels also include targets in foreign countries that are liquid and that DOE expects would be converted to oxide prior to shipment to SRS. Only about 10 percent of the Group D fuel is in storage at SRS. The rest of the material has yet to be produced via foreign research reactor operations. Although eligible for shipment, most of this fuel is not part of the current shipping plan as projected by foreign research reactor operators.

The Group D fuel currently stored at SRS (676 cans of Sterling Forest Oxide fuel from the former medical isotope – production reactor; see Table C-4) presents a potential health and safety vulnerability similar to that of the Group A fuels. If a storage can leaked, DOE would address the problem as described for the Group A fuels to prevent the release of fission products and particulate matter to the water of an SRS storage basin. Group D comprises approximately 6 percent of the volume of the aluminum-based SNF that DOE could manage at SRS from now until 2035.

Higher Actinide Targets (Group E):

This group contains irradiated and unirradiated target materials used to generate radionuclides with atomic numbers higher than that of uranium. This material could be used to support such national programs as space exploration or medical research. The targets are aluminum-clad plutonium oxide that contain significant quantities of americium and curium, which react under neutron irradiation to produce elements with still higher atomic numbers such as californium. All materials in this group are stored in the Receiving Basin for Offsite Fuel. Group E accounts for less than 1 percent of the volume of aluminum-based SNF DOE could manage at SRS from now until 2035.

The Higher Actinide Target fuel group consists of 60 Mark-51 targets, 114 other targets, and 65 Mark-18 targets. This material was evaluated in the *Final Environmental Impact Statement for Interim Management of Nuclear Materials*, (DOE/EIS-0220) and DOE decided the targets should remain in wet storage. In this EIS, DOE evaluates the continued wet storage of the Mark-51 and other targets pending shipment offsite. DOE also evaluates repackaging the Mark-51 and other targets to place them in a new dry storage facility so that the material could be transferred to dry storage if necessary to provide flexibility in spent fuel storage operations.

The Mark-18 targets are different from the Mark-51 and other targets in several ways. The most important distinction is that each Mark-18 target is one continuous piece about 12 feet long. The Mark-51 and other targets are about 2 feet long. The Mark-51 and other targets could be handled, transported and stored (including in a dry storage facility) in their current configuration. The 12-foot long Mark-18 targets would require size reduction for transport or storage in a dry storage facility. The standard method to reduce the size of the Mark-18 targets would be to cut them up under water in an SRS wet storage basin. The condition of the Mark-18 targets presents a health and safety vulnerability for under water cutting because of the suspected brittle condition of the targets and the uncertainty of the region of the target assemblies that contains the target product (i.e., americium and curium) and fission products. The brittle condition is due to a very long irradiation cycle in a reactor at the SRS. Cutting the targets using the existing site capability could result in the uncontrolled

release of radioactive material to the water of the Receiving Basin for Offsite Fuel. For these reasons, a previous DOE assessment of this material (see Section 1.6.2) concluded that the Department should consider processing the Mark-18 targets in F Canyon. Analysis of such alternatives are not included in this EIS because DOE performed that evaluation in the *Final Environmental Impact Statement for Interim Management of Nuclear Materials*, which is incorporated herein by reference. Those alternatives included dissolving the targets in F-Canyon and then vitrifying the americium and curium in a new F-Canyon vitrification facility, dissolving the targets in F-Canyon and recovering the americium and curium as an oxide, and dissolving the targets and transferring the americium and curium to the high-level waste tanks at the SRS.

Non-Aluminum-Clad Fuels (Group F):

This group consists of the large variety of stainless-steel or zirconium-clad SNF at SRS that DOE plans to ship to INEEL in accordance with decisions DOE reached under the SNF PEIS (DOE 1995b).

1.5.1 COMPARISON OF SPENT NUCLEAR FUEL GROUPS

A comment was made regarding the differences between the fuel categories used in this EIS and the EIS for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada (i.e., Yucca Mountain EIS). The Notice of Availability of the Yucca Mountain Draft EIS was published on August 13, 1999 (64 FR 44217) and analyzes the options being considered for siting of a repository for spent nuclear fuel and high level waste.

Table 1-2 shows the categories being used in both EISs. The Yucca Mountain categories and MTHM numbers encompass fuel and targets being managed by SRS in preparation for ultimate disposition. Should a repository be developed, that fuel and most targets would be shipped, in one form or another, to the repository for ultimate disposition. Category F fuel will be shipped from SRS to INEEL under the Record of Decision for the Final Programmatic Spent Nuclear Fuel and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs EIS. As such, INEEL will be responsible for determining the ultimate disposition of category F fuel. Therefore, the 20.4 MTHM of non-aluminum clad fuel is not included in the Yucca Mountain categories for SRS managed fuel.

Category A is made up of 17 MTHM EBR-II (matching Yucca Mountain EIS category 1) and 2 MTHM SRE ("Thorium" part). The SRE is contained within Yucca mountain category 16.

Material within groups B and C of the SRS SNF EIS are included in groups 5, 6, and 7 of the Yucca Mountain EIS. Material within groups D & E of the SNF EIS are included in group 16 of the Yucca Mountain EIS. The material is made up of foreign research reactor and domestic research reactor fuel and targets and other target material produced at SRS.

Excluding group F, there is a 4.0 MTHM difference between the totals calculated for the SNF EIS table (47.8 MTHM) and the Yucca Mountain table (43.8 MTHM). The differences are due to recent decisions by some foreign research reactor (FRR) operators which have reduced the quantity of SNF expected to be shipped to SRS. However, the SRS SNF EIS uses the larger projected number because those FRRs still have the option to ship to the United States.

Table 1-2. Comparison of Spent Nuclear Fuel Groups.

NEPA document		Fuel group	Mass (MTHM)[a]
Savannah River Site Spent Nuclear Fuel Management EIS (DOE/EIS-0279)	A	Uranium and Thorium Metal Fuels	19
	B	Material Test Reactor-Like Fuels	20
	C	HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	8 0.7
	D	Loose Uranium Oxide	0.1
	E	Higher Actinide Targets	20.4
	F	Non-Aluminum-Clad Fuels	
Draft EIS for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, NYE County, Nevada (DOE/EIS-0250D)[b]	1	Uranium Metal	17
	5	Uranium Oxide, Failed/ Declad/ Aluminum Clad	3.2
	6	Uranium-Aluminide	8.7
	7	Uranium-Silicide	12
	16	Miscellaneous	2.9

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a. MTHM = Metric tons of heavy metal.
b. Includes only Savannah River Site Fuel

1.6 Relevant Documents

1.6.1 NATIONAL ENVIRONMENTAL POLICY ACT DOCUMENTS

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

DOE prepared this EIS (DOE 1995b) in compliance with a Court Order dated December 22, 1993, in the case of Public Service Company of Colorado v. Andrus, No. 91-0054-5-HLR (D. Idaho). The preferred alternative in the Final EIS, which DOE issued in April 1995, is Regionalization by Fuel Type. Volume 1 of this EIS analyzes at a programmatic level potential environmental impacts over the next 40 years of alternatives related to the transportation, receipt, processing, and storage of DOE-owned SNF. Volume 1 supports programmatic decisions on sites at which DOE will manage various types of SNF.

In the Record of Decision, which selected the preferred alternative for implementation (60 FR 28680), DOE decided to manage its SNF by type (fuel cladding and matrix material) at the Hanford Site, the Idaho National Engineering and Environmental Laboratory, and the SRS. Section C.1.2 in Appendix C of this SRS SNF Management EIS discusses its relationship to the programmatic SNF EIS.

An amendment to the Record of Decision (61 FR 9441) reflects the October 16, 1995, Settlement Agreement between DOE, the State of Idaho, and the Department of the Navy by reducing the number of proposed spent fuel shipments to Idaho.

Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor SNF

This EIS (DOE 1996a) analyzes the management of foreign research reactor SNF that contains uranium originally produced or enriched in the United States. It also analyzes appropriate ways to manage such fuel received in the United States, amounts of fuel, shippers, periods of time over which DOE would manage the fuel, modes of transportation, and ownership of the fuel. In

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its Record of Decision (61 FR 25091), DOE stated it would accept from 41 listed countries aluminum-based spent fuel, Training Research Isotope General Atomic (TRIGA) spent fuel, and target material containing uranium enriched in the United States. L2-5

EC Over the life of the foreign research reactor SNF acceptance program, DOE could accept approximately 19.2 MTHM of foreign research reactor SNF in as many as 22,700 separate elements and approximately 0.6 MTHM of target material. Most of the fuel will arrive through the Charleston Naval Weapons Station in South Carolina (about 80 percent), with a very limited amount arriving through the Concord Naval Weapons Station in California (about 5 percent). Most of the target material and some of the fuel (about 15 percent) will arrive overland from Canada. Shipments through Charleston began in September 1996 and those through Concord began in July 1998.

TC After a limited period of storage, DOE will process and package the fuel as necessary at the SRS and the Idaho National Engineering and Environmental Laboratory to prepare it for disposal in a geologic repository. Section C.1.2 in Appendix C explains the relationship of the Foreign Research Reactor SNF EIS to this EIS.

Final Environmental Impact Statement Interim Management of Nuclear Materials

This EIS (DOE 1995a) evaluates actions to stabilize SRS materials that represent environmental, safety, and health vulnerabilities in their current storage condition or that might represent a vulnerability within the next 10 years.

EC DOE has published four decisions under this EIS. In the first (60 FR 65300), DOE decided to process plutonium-242 solutions to oxide; vitrify americium and curium solutions to glass; blend highly-enriched uranium solutions down to low enrichment; process the plutonium in Mark-31 target slugs; process plutonium and uranium material in vaults to metal, oxide, or glass, if necessary; and process failed Taiwan Research Reactor SNF and a failed canister of Experimental Breeder Reactor-II SNF.

DOE decided that processing the EBR-II fuel in unbreached canisters was not immediately necessary. EBR-II fuel is declad and reactive, but only when it is in contact with water. The fuel inside a storage canister will not corrode as long as the canister retains its integrity. A monitoring and inspection program is in place that would detect any change in the integrity of the storage canisters. Any canisters that failed would be detected and the fuel then processed under the provisions of the Record of Decision to stabilize the material. This monitoring and inspection program applies as well to other fuel types in storage.

In the first supplement to the Record of Decision (61 FR 6633), DOE decided to stabilize Mark-16 and -22 fuels by processing them in the SRS canyons and blending the resulting highly enriched uranium down to low enriched uranium; and to stabilize "other aluminum-clad targets" by dissolving them in the canyons. DOE will transfer the resulting nuclear material from the targets to the SRS high-level waste tanks for vitrification in the Defense Waste Processing Facility.

The second supplement to the Record of Decision (61 FR 48474) contains decisions on vitrifying neptunium-237 solutions, and on the stabilization of plutonium-239 solutions by converting them to a metal using the F and H Canyons and FB-Line.

In the third supplement to the Record of Decision (62 FR 17790), DOE decided to use the F Canyon and FB-Line to stabilize the remaining Taiwan Research Reactor SNF in the Receiving Basin for Offsite Fuel. These actions are relevant to the cumulative impacts assessment in this EIS (see Chapter 5).

Disposition of Surplus Highly Enriched Uranium Environmental Impact Statement

DOE prepared this EIS (DOE 1996b) because of the need to reduce the threat of nuclear weapons proliferation worldwide in an environmentally TC

safe manner by reducing stockpiles of weapons-usable fissle materials, setting a non-proliferation example for other nations, and allowing peaceful, beneficial use of the material to the extent practical.

In the Record of Decision (61 FR 40619), DOE stated it would implement a program that will gradually blend as much as 85 percent of the surplus highly enriched uranium to a uranium-235 enrichment level of approximately 4 percent, and will blend the remaining surplus highly enriched uranium down to an enrichment level of about 0.9 percent for disposal as low-level waste. This will occur over 15 to 20 years. DOE could use different technologies at four potential blending facilities, including SRS and the Oak Ridge Reservation. Blending down of highly-enriched uranium would affect SRS operations and waste generation. This activity is relevant to the assessment of cumulative impacts (see Chapter 5).

Storage And Disposition Of Weapons-Usable Fissile Materials Programmatic Environmental Impact Statement

DOE prepared this programmatic EIS (DOE 1996c) to evaluate a safe and secure strategy for the long-term storage of weapons-usable fissile materials, primarily plutonium-239 and highly enriched uranium, and the disposition of weapons-usable plutonium that was surplus to national defense needs. This EIS included the SRS inventory of plutonium-239, highly enriched uranium, and other weapons-usable materials.

The Record of Decision (62 FR 3014) specified that DOE will expand or upgrade SRS facilities (i.e., the Actinide Packaging and Storage Facility) to consolidate weapons-usable plutonium, and will move plutonium pits now stored at the Rocky Flats Environmental Technology Site in Colorado to the Pantex Plant in Texas and non-pit plutonium materials to SRS. DOE will ship the non-pit plutonium to SRS only if a subsequent decision calls for the immobilization of plutonium at the Site. The DOE disposition strategy enables the immobilization of surplus plutonium in glass or ceramic material for disposal in a geologic repository, and the burning of some surplus plutonium as mixed oxide fuel in domestic commercial reactors with subsequent disposal of the spent fuel in a geologic repository in accordance with the Nuclear Waste Policy Act.

DOE specified that it will determine the exact locations for disposition of these materials in site-specific EISs and in cost, technical, and nonproliferation studies. However, DOE has decided that it will locate a vitrification or immobilization facility (with a plutonium conversion facility) at either the Hanford Site in Washington or SRS, and that SRS is a candidate site for a potential mixed oxide fuel fabrication facility and a pit disassembly and conversion facility. The implementation of these decisions will require several years. The Programmatic Weapons-Usable Fissile Materials EIS is also relevant in the assessment of cumulative impacts that could occur at the SRS (see Chapter 5).

The Department issued an Amended Record of Decision (63 FR 43386) to the environmental impact statement, *Storage and Disposition of Weapons-Usable Fissile Materials*, on August 6, 1998. In order to support the early closure of the Rocky Flats Environmental Technology Site (RFETS) and the early deactivation of plutonium storage facilities at the Hanford Site, DOE modified, contingent upon the satisfaction of certain conditions, some of the decisions made in its Storage and Disposition ROD associated with surplus plutonium storage pending disposition. Namely, DOE will take steps that allow: (1) the accelerated shipment of all non-pit surplus weapons-usable plutonium from the RFETS (about 7 metric tons) to the SRS beginning in about 2000, in advance of completion of the Actinide Packaging and Storage Facility in 2001, and (2) relocation of all Hanford surplus weapons-usable plutonium (about 6.4 metric tons) to the SRS, between about 2002 and 2005, pending disposition. However, consistent with the Storage and Disposition PEIS ROD, DOE will only implement the movement of the RFETS and Hanford plutonium inventories to the SRS if the

SRS is selected as the immobilization disposition site. DOE is preparing the *Surplus Plutonium Disposition EIS*, draft issued July 1998, as part of the decision-making process for determining the immobilization site. The action described in this EIS is relevant in the assessment of cumulative impacts that could occur at SRS (see Chapter 5).

Final Defense Waste Processing Facility Supplemental Environmental Impact Statement

DOE prepared a Supplemental EIS to examine the impacts of completing construction and operating the Defense Waste Processing Facility at the SRS. This document (DOE 1994) assisted the Department in deciding whether and how to proceed with the Defense Waste Processing Facility project, given the changes to processes and facilities that had occurred since 1982, when it issued the original Defense Waste Processing Facility EIS. The Record of Decision (60 FR 18589) announced that DOE would complete the construction and startup testing of the Defense Waste Processing Facility, and would operate the facility using the In-Tank Precipitation process after the satisfactory completion of startup tests.

The alternatives evaluated in this EIS on the management of SNF could generate radioactive waste that DOE would have to handle or treat at facilities described in the Defense Waste Processing Facility Supplemental EIS and the SRS Waste Management EIS (see next paragraph). The Defense Waste Processing Facility Supplemental EIS is also relevant to the assessment of cumulative impacts (see Chapter 5) that could occur at SRS.

Savannah River Site Waste Management Final Environmental Impact Statement

DOE issued the SRS Waste Management EIS (DOE 1995c) to provide a basis for the selection of a sitewide approach to managing present and future (through 2024) wastes generated at SRS. These wastes would come from ongoing operations and potential actions, new missions, environmental restoration, and decontamination and decommissioning programs.

The SRS Waste Management EIS includes the treatment of wastewater discharges in the Effluent Treatment Facility, F- and H-Area tank operations and waste removal, and construction and operation of a replacement high-level waste evaporator in the H-Area tank farm. In addition, it evaluates the Consolidated Incineration Facility for the treatment of mixed waste. The Record of Decision (60 FR 55249) stated that DOE will configure its waste management system according to the moderate treatment alternative described in the EIS. The SRS Waste Management EIS is relevant to this SNF Management EIS because it evaluates management alternatives for various types of waste that actions proposed in this EIS could generate. The Waste Management EIS is also relevant in the assessment of cumulative impacts that could occur at the SRS (see Chapter 5).

Environmental Impact Statement for a Geologic Repository for the Disposal of SNF and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada

On August 13, 1999, DOE announced the availability (64 FR 44200) of a draft environmental impact statement for a geologic repository at Yucca Mountain for the disposal of SNF and high-level radioactive waste, in accordance with the Nuclear Waste Policy Act of 1982. The DEIS evaluates site-specific environmental impacts from the construction, operation, and closure of the repository. It also evaluates reasonable alternatives for implementing such a proposal, and transportation-related impacts for shipments from across the United States. The DEIS also evaluates the consequences at SRS of continued SNF and high-level waste management assuming the repository is not constructed and operated. The repository decision will affect the ultimate disposal of SNF from SRS. The Final EIS is scheduled to be completed in Fiscal Year 2001.

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Treatment and Management of Sodium-Bonded Spent Nuclear Fuel Environmental Impact Statement

DOE has published a draft environmental impact statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel (64 FR 8553 2/22/99). Alternatives to processing at the Idaho National Engineering and Environmental Laboratory (INEEL) include the use of the Plutonium-Uranium Extraction (PUREX) solvent extraction method currently in use at SRS and the melt and dilute technology that is being proposed under this EIS. The technologies would be applied to sodium-bonded spent nuclear fuel blanket assemblies, which are currently in storage at INEEL. L2-16 There is approximately 22.4 MTHM of Experimental Breeder Reactor-II (EBR-II) blanket fuel and 34.2 MTHM of Fermi-1 blanket fuel to be processed. This EIS includes cumulative impacts of sodium-bonded spent nuclear fuel processing at the SRS based on estimates from conventional processing of Fuel Group A. Fuel Group A is mostly EBR-II fuel (16.7 MTHM out of 19 MTHM) and therefore provides a good basis for estimating impacts from processing of similar material at SRS. DOE estimates that the impacts for conventional processing would be sufficiently representative of impacts from melt and dilute for the purpose of presenting cumulative impacts.

Management of Certain Plutonium Residues and Scrub Alloy at the Rocky Flats Environmental Technology Site Final Environmental Impact Statement

In August 1998, the Department issued the Final EIS (DOE 1998a). In this EIS DOE proposed to process certain plutonium-bearing materials being stored at the Rocky Flats Environmental Technology Site (Rocky Flats) located near Golden, Colorado. These materials are plutonium residues and scrub alloy remaining from nuclear weapons manufacturing operations formerly conducted by DOE at that site. In their present forms, these materials cannot be disposed of or otherwise dispositioned because they contain plutonium in concentrations exceeding DOE safeguards termination requirements.

DOE has decided to ship approximately 7,450 pounds of sand, slag and crucible and plutonium fluoride residues (containing approximately 600 pounds of plutonium) and approximately 1,543 pounds of scrub alloy (containing L2-16 approximately 440 pounds of plutonium) to SRS where these materials will be stabilized in F Canyon by chemically separating the plutonium from the remaining materials in the residues and scrub alloy. The separated plutonium will be placed in safe and secure storage, along with a larger quantity of plutonium already in storage at the Savannah River Site, until DOE has completed the *Surplus Plutonium Disposition Environmental Impact Statement* and made final decisions on the disposition of the separated plutonium. Transuranic wastes generated during the chemical separations will be sent to the Waste Isolation Pilot Plant for disposal. Other wastes generated during the chemical separations operations will be disposed of in accordance with the Savannah River Site's normal procedures for disposing of such wastes. The actions will occur between 1998 and 2002.

Final Environmental Impact Statement Accelerator Production of Tritium at Savannah River Site **(DOE, 1998b)** EC

DOE has proposed an accelerator design (using helium-3 target blanket material) and an alternate accelerator design (using lithium-6 target blanket material). If an accelerator is built, it would be located at SRS. In the Record of Decision DOE decided to use an existing commercial light-water TC reactor as the new tritium source. Therefore, the accelerator will not be built at SRS and impacts from construction and operation are not included in the cumulative impacts section of this EIS.

Final Environmental Impact Statement for the Construction and Operation of a Tritium Extraction Facility at the Savannah River Site **(DOE 1998c)** EC EC

As stated in the Record of Decision (64 FR 26369; 5/14/99), DOE will construct and operate a Tritium Extraction Facility on SRS to provide the capability to extract tritium from commercial TC

light water reactor targets and targets of similar design. The purpose of the proposed action and alternatives evaluated in the EIS is to provide tritium extraction capability to support either accelerator or reactor production. The Tritium Extraction Facility EIS is relevant in the assessment of cumulative impacts that could occur at SRS (see Chapter 5).

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***Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999)**

This EIS analyzes the activities necessary to implement DOE's disposition strategy for surplus plutonium. Following completion of the EIS, SRS was selected (65.FR 1608) as the location for mixed oxide fuel fabrication and plutonium immobilization facilities that would be used for plutonium disposition, and for the plutonium pit (a component of nuclear weapons) disassembly and conversion facility. The projected impacts of these operations are incorporated in Chapter 5 of this EIS.

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1.6.2 OTHER RELEVANT DOCUMENTS

In August 1997, DOE chartered the Nuclear Materials Processing Needs Assessment. The purpose of the assessment was to determine which, if any, additional nuclear materials within the Department of Energy complex may require use of the SRS chemical separations facilities (F or H canyon) for stabilization or preparation for disposition prior to canyon de-commissioning. Chemical separations operations are occurring at SRS because DOE is using the canyons to stabilize nuclear materials that represent potential health and safety risks in their current storage configuration. The decisions to use processing capabilities have been documented in a number of Records of Decision, including those following the *F-Canyon Plutonium Solutions EIS*, the *Interim Management of Nuclear Materials EIS*, and the *Rocky Flats Plutonium Residues EIS*. These decisions are consistent with DOE's Implementation Plan for Defense Nuclear Facilities Safety Board Recommendation 94-1, wherein the Board recommended that DOE take steps, including use of the processing facilities, to stabilize nuclear materials that represented health and safety risks.

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The Processing Needs Assessment evaluated four material categories that could require the canyons for stabilization or disposition: spent nuclear fuel, plutonium-239, uranium, and other special isotopes. The results of the assessment are being reviewed by DOE management to identify needed follow-on actions.

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Other materials under consideration for processing as SRS canyons include various components currently located at other DOE sites, including Oak Ridge, Rocky Flats, Los Alamos, and Hanford. These materials, which were identified during the Processing Needs Assessment, consist of various plutonium and uranium components. If DOE were to process these materials in the SRS separations facilities, additional NEPA reviews would need to be performed. This material has been considered in the cumulative impacts presented in Chapter 5.

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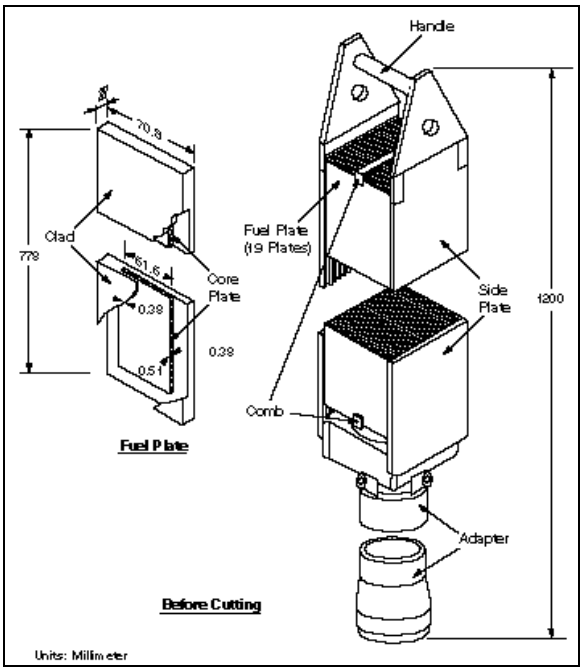

Figure 1-3. Typical Materials Test Reactor fuel assembly.

Although some Group B fuels are stored at SRS in the Receiving Basin for Offsite Fuel or in L Disassembly Basin, at present most are at domestic universities, foreign research reactors, and DOE research facilities pending shipment to the Site. All of the Group B fuels that are currently stored at SRS are "intact." The good condition of the cladding and the durability of the alloyed fuel at SRS provide a high degree of confidence that the fuel will not degrade during storage and that actions to correct potential health and safety vulnerabilities will not be necessary before treatment using the technology that DOE proposes to select under the record of decision from this EIS. DOE expects this will be true for most of the foreign and domestic research reactor SNF included in Group B that is yet to be shipped to SRS. However, if DOE determines that any of the Group B fuel presents a health and safety vulnerability, DOE would evaluate the situation and take appropriate action that could include canning the problem fuel or processing the fuel in one of the SRS canyon facilities. This management approach is consistent with the Record of Decision reached under the *Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel.*

HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (Group C):

Fuels in this group are similar in composition to Group B fuels in that they are aluminum-based, highly enriched uranium (HEU) and low enriched uranium (LEU) oxides and silicides, but their size or shape might preclude packaging them in the disposal canisters proposed for use in a repository without resizing or special packaging considerations. Some fuel in this group is smaller in diameter and longer than Group B fuels or is larger than Group B fuels in both diameter and length; it often comes in odd shapes such as a 1.5-foot by 3-foot (0.46-meter by 0.9-meter) cylinder or a sphere with a diameter of 29 inches (74 centimeters). DOE would have to disassemble or use other volume-reduction activities to place such fuels in a nominal 17-inch direct co-disposal canister (see Section 2.2). At present, much of this fuel is at other DOE sites and in other countries but is scheduled to be received at SRS. EC EC

DOE expects that most of the fuel in this category is intact and would be managed as described above for Group B fuels. However, a small amount is not intact. That material consists of some fuel and one target that were cut or sectioned for research purposes. After the research was completed, the fuel and target pieces were canned in 14 cans and placed in wet storage. The origin and location of this material is discussed in Appendix C, Table C-3. The sectioned fuel and target present a potential health and safety vulnerability similar to that of the Group A fuel discussed previously. If a storage can were to leak, DOE would address the problem as described for the Group A fuel to prevent the release of fission products and particulate material to the water of a storage basin. Additionally, the current form of the fuel (i.e., failed) may not be acceptable in a repository because its integrity has been compromised. EC

CHAPTER 2. PROPOSED ACTION AND ALTERNATIVES

This chapter describes the U.S. Department of Energy's (DOE) proposed action; that is, the management of spent nuclear fuel (SNF) at the Savannah River Site (SRS). Technical terms are defined in the Glossary.

2.1 Proposed Action

As described in Chapter 1, SRS will receive aluminum-based SNF from foreign research reactors, domestic research reactors, and other DOE sites. DOE will have to manage this fuel, in addition to some SNF already stored at the Site, in a manner that will protect human health and the environment. Additionally, DOE is committed to avoiding indefinite storage at SRS of SNF that is in a form unsuitable for final disposition. Therefore, DOE's proposed action is to safely manage SNF that is currently located or expected to be received at SRS, including treating or packaging aluminum-based SNF for possible offsite shipment and disposal in a geologic repository, and packaging non-aluminum clad fuel for on-site dry storage or offsite shipment.

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TC

In the Record of Decision (ROD) for the Final Environmental Impact Statement on a Proposed Nuclear Nonproliferation Policy Concerning Foreign Research Reactor SNF (61 FR 25092), DOE stated that it would embark on an accelerated program at SRS to identify, develop, and demonstrate one or more non-chemical processing, cost effective treatment or packaging technologies to prepare aluminum-based foreign research reactor spent nuclear fuel for ultimate disposition.

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Based on that decision, DOE's proposal is to select a new non-chemical processing technology that would put aluminum-based foreign research reactor SNF into a form or container suitable for direct placement in a geologic repository. Treatment or conditioning of the fuel would address potential repository acceptance criteria and potential safety concerns. Implementing the new non-chemical processing treatment or packaging technology would allow DOE to manage the SNF in a road-ready condition at SRS in dry storage pending shipment offsite.

TC

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Because of the similarity of the material, DOE proposes to manage the other aluminum-alloy SNF that is the subject of this EIS (domestic research reactor and DOE reactor fuels) in the same manner as the foreign research reactor fuels.

In the Final Environmental Impact Statement on a Proposed Nuclear Nonproliferation Policy Concerning Foreign Research Reactor SNF Record of Decision, DOE stated that, should it become apparent by the year 2000 that DOE will not be ready to implement a new SNF treatment technology, DOE would consider chemically processing foreign research reactor SNF in F Canyon. The Final Environmental Impact Statement on a Proposed Nuclear Nonproliferation Policy Concerning Foreign Research Reactor SNF Record of Decision described the possible use of F Canyon for SNF processing based on a preliminary concept to consolidate all processing operations in one canyon. Subsequent review has shown that consolidating highly enriched uranium spent fuel processing operations in F Canyon would not be practical due to criticality considerations and process capacity restrictions associated with the plutonium-uranium extraction system used in F Canyon. Thus, DOE is now proposing to use H Canyon to chemically separate highly enriched uranium spent fuel.

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DOE also committed that any decision to use conventional chemical processing would consider the results of a study (62 FR 20001) on the nonproliferation, cost, and timing issues associated with chemically processing the fuel. DOE stated that any highly enriched uranium separated during chemical processing would be blended down to low enriched uranium.

DOE has included chemical processing as a management alternative in this EIS, although

DOE's preference is to use non-chemical operations processes. DOE proposes to use conventional processing to stabilize some materials before a new treatment facility is in place. The rationale for this is to avoid the possibility of urgent future actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

EC The limited proposed canyon processing actions is not expected to extend the operating schedules for these facilities beyond the current planning basis. Processing would eliminate potential health and safety vulnerabilities that could occur prior to the availability of a new SNF treatment technology. In the event a new treatment process becomes available, the SNF with potential health and safety vulnerabilities could be processed using the new treatment technology.

EC Previous DOE management decisions on disposition of SNF are outlined in Section 1.1 and Appendix C, Section C.1.2. Relevant National Environmental Policy Act documents are discussed in Section 1.6.

EC 2.2 Spent Nuclear Fuel Management Technology Options

DOE has identified 11 potential treatment and packaging technology options in addition to conventional processing that could be used to prepare aluminum-based SNF at SRS for final disposition in a geologic repository. All of the technology options are discussed in Appendix A of this EIS.

TC

Two of the options, Direct Disposal and Direct Co-Disposal, are non-destructive methods to prepare and package aluminum-based SNF for disposition in a geologic repository. Another technology option, Repackage and Prepare to Ship, is pertinent only to non-aluminum-clad SNF and programmatic material that would be shipped offsite. These three technology options are discussed under the New Packaging Technology options section (Section 2.2.3) of this EIS.

Nine of the technology options are potential processes for the treatment of aluminum-based SNF. These are Melt and Dilute, Press and Dilute, Chop and Dilute, Plasma Arc Treatment, Glass Material Oxidation and Dissolution System, Dissolve and Vitrify, Electrometallurgical Treatment, Can-in-Canister, and Chloride Volatility. DOE has consolidated seven of these processing technology options into four categories for analysis in this EIS. The Press and Dilute and the Chop and Dilute options are similar, so DOE has represented them for analysis as Mechanical Dilution. The Plasma Arc Treatment, the Glass Material Oxidation and Dissolution System, and the Dissolve and Vitrify options use processes that produce a product with properties similar to that produced at the Defense Waste Processing Facility (DWPF) at SRS. Therefore, DOE has represented these three as the Vitrification option. The Melt and Dilute and the Electrometallurgical Treatment options are analyzed separately. The new treatment options are discussed under the New Processing Technology section of this EIS (Section 2.2.4).

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DOE considered the remaining two technology options but dismissed them from analysis in this EIS. With Chloride Volatility, SNF would react with chlorine gas at high temperatures to form volatile chlorides. The uranium, aluminum, fission products, and transuranics would be separated from each other by cooling and distillation. This technology is very immature

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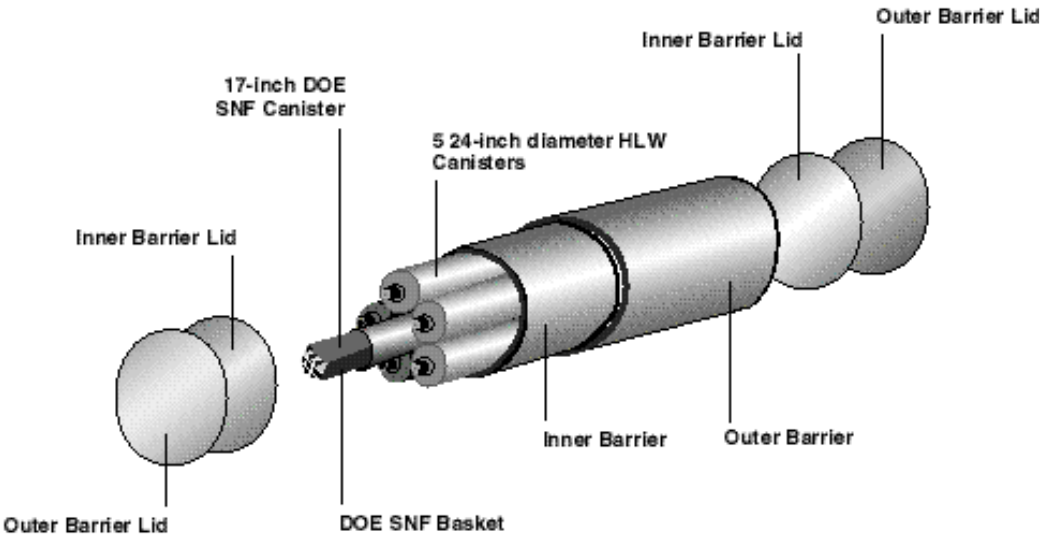
Co-Disposal packaging – strategy for all options requiring shipment to the geologic repository



Two alternatives, New Packaging Technology and New Processing Technology, would result in the dry storage of SNF in a road-ready condition. Under these alternatives, the fuel would be contained in stainless-steel canisters. At the repository the canisters would be loaded into a repository waste package with canisters of vitrified high-level waste. DOE expects five canisters of high-level waste would fit in a repository waste package, leaving room for one canister of SNF. This approach is termed Co-Disposal. It would enable repository disposal of SNF with no space requirements beyond those needed for the disposal of the high-level vitrified waste. The high radiation field of the vitrified high-level waste would provide safeguards protection against unauthorized diversion of the SNF for recovery of the enriched uranium.

The SNF canisters would be packaged for Co-Disposal with high-level vitrified waste canisters at the reposi-

in terms of actual development and testing and the potential for implementation in a timely manner is very uncertain. In addition, this method of chemical separation offers no advantage over conventional processing and DOE eliminated the option from further consideration.

The second technology option dismissed from analysis was Can-in-Canister, under which DOE would place SNF in a can (in an amount that would not pose criticality concerns), place the can in a stainless-steel canister, and fill the canister with vitrified high-level waste. This technology was originally developed as a means for disposing of immobilized plutonium. Because plutonium does not emit intense penetrating radiation, the high radiation field of the vitrified high-level waste would render the plutonium inaccessible. However, a more cost-effective and technologically viable way to protect the SNF with radiation fields is to employ the co-disposal concept. Should the Can-in-Canister method be used with aluminum SNF, the high temperature of the molten glass could melt the aluminum in the fuel, changing the geometry of the fuel matrix in an uncontrolled fashion. Therefore, this option could pose significant risks to human health and the environment, and for that reason was not considered a reasonable alternative. TC

The New Packaging Technology options and the New Processing Technology options consist of several technology options that DOE has not previously applied to the management of aluminum-based SNF for the purpose of ultimate disposition. As a result, DOE believes that the highest confidence of success and greatest technical suitability lies with options that have relatively sim- EC

ple approaches (i.e., Direct Disposal, Direct Co-Disposal, Melt and Dilute, and Press and Dilute).

2.2.1 REPOSITORY CONSIDERATIONS

EC As discussed in Section 2.1, part of DOE's proposed action is to prepare SNF to meet the requirements that the Department anticipates will be applicable to material to be placed in a geologic repository. Any technology that DOE implements must be able to provide a product that is compatible with such criteria. DOE must rely on reasonable assumptions about what the acceptance criteria would include when making decisions on SNF treatment technologies. As described in Chapter 1, DOE anticipates that TC eventually it will place its aluminum-based SNF inventory after treatment or repackaging in a geologic repository.

EC As the operator of any geologic repository for SNF, DOE would be responsible for developing acceptance criteria for the material that would be TC placed in the repository. However, the U.S. Nuclear Regulatory Commission (NRC) would be EC responsible for licensing the repository. Therefore, DOE is working closely with the NRC to develop acceptance criteria. DOE will provide EC the NRC with characterization data for material that would be prepared for disposal in a geologic EC repository. At this time, acceptance criteria need to be conservative because of uncertainties concerning any engineered or natural barriers at a repository. However, as repository and packag- TC ing designs evolve, the criteria will become more detailed. Fuel characterization data will need to be detailed enough to verify that each element or canister falls within the ultimate acceptance criteria. Such detail, however, is not currently available. Final acceptance criteria will not be available until after NRC issues its authorization, TC based on the successful demonstration of safe, long-term performance of the candidate repository in accordance with NRC regulations. Until such time, the preliminary acceptance criteria tend to be conservative to allow for uncertainties in performance of engineered or natural barriers and how such performance may impact public and worker health and safety, and material isolation.

DOE has performed preliminary evaluations of the expected SNF characteristics (DOE 1995a, 1996a). Those evaluations indicated that the SNF to be placed in the repository would have to meet requirements for the following characteristics:

Packaging

- Dimension and weight limits
- Material compatibility
- Thermal limits
- Internal gas pressure limits
- Labeling
- Handling ability
- Waste isolation

Contents

- Solid material – no particulates
- Noncombustible
- No free liquids
- No hazardous waste (as defined by the Resource Conservation and Recovery Act)

Chemical reactivity

- Not chemically reactive
- Nonpyrophoric
- Nonexplosive

Nuclear material safeguards

- Reduced uranium-235 enrichment
- Self-protecting radiation fields
- Tamper-proof seals

Criticality control

- Limits on nuclear reactivity by controlling amount of uranium and its enrichment (see Text Box on page 2-5)

Proliferation and Criticality Concerns for SNF Disposal

Preparation of SNF for disposal in a geologic repository requires consideration of the risk of a disruptive nuclear criticality. Criticality risk is defined as the potential for a neutron-induced self-sustaining fission reaction like that which occurs in a nuclear reactor. Nuclear criticality in the SNF would be due to uranium enriched in the fissile nuclide uranium-235 with the remainder being principally non-fissile uranium-238. Characteristic enrichment levels in these fuels are designated as follows (DOE 1996b).

	Percent uranium-235
Highly enriched uranium (HEU)	>20-93
Low enriched uranium (LEU)	>2-≤20
Commercial power reactor fuel	<2-4
Very low enriched uranium (VLEU)	≤2
Natural uranium (NU)	0.72
Depleted uranium (DU)	Typically 0.18

Concern for the enrichment level of the fuel arises from two considerations: (1) weapons material proliferation policy and (2) criticality control during storage, transportation, and repository disposal. The high-enriched uranium fuels are generally considered to present unacceptable proliferation risks, unless otherwise protected. Isotopic dilution of the high-enriched uranium fuels to 20 percent uranium-235 during treatment for repository disposal satisfies requirements for protection against this proliferation risk.

TC One approach to control the potential for a nuclear reaction during storage, transport, and repository disposal of the SNF (high-enriched uranium or low-enriched uranium) is addressed by incorporation of neutron-absorbing poison materials in the waste form or containers, by reduction of enrichment levels to the extent practical (2 to 20 percent), and by limiting the mass loading of fissile uranium-235 in the primary waste form canisters. Provisional limits for fissile mass loadings have been specified as follows (DOE 1996b):

	Allowable fissile mass loading (kg U-235) per canister*
HEU	14.4
LEU	43
VLEU	200

*Larger quantities of fissile U-235 in the canister are permitted at lower enrichment levels because of neutron escape or absorption in non-fissile material.

In accord with these specifications, the SNF processed for Direct Co-Disposal (with no dilution of highly enriched uranium) would require incorporation of neutron poisons in the waste canister and possibly smaller canisters to meet fissile mass loading limits. The processes under the New Processing Technology, which would achieve enrichment levels of 20 percent or less, would generate canisters within the low-enriched uranium fissile mass loading limits, but could require incorporation of poison materials for additional criticality

Radiation

- Radiation field limits
- Canister surface contamination limits

L5-3 The preliminary acceptance criteria describe the physical, chemical, and thermal characteristics to which spent nuclear fuel, high-level waste, and associated disposable canisters must conform for emplacement in the repository. The preliminary criteria are organized into four categories:

- General/Descriptive
- Physical/Dimensional L5-3
- Chemical/Compatibility
- Thermal/Radiation/Pressure

Disposability Assessment: Aluminum-Based Spent Nuclear Fuel Forms (WSRC 1998a) provides a technical assessment of the Melt-and-Dilute and Prepare for Direct Disposal/Direct Co-Disposal technologies against these preliminary criteria. This assessment is based on results of several analytical and experimental –investigations at SRS, and criticality calculations. The Disposability Assessment concluded:

> Both Melt-Dilute and Direct [disposal] forms [for aluminum-alloy SNF] in disposable containers can meet the requirements of the Draft Standards for Spent Nuclear Fuel in Disposable Canisters. Completed analyses indicate that the Melt-Dilute form of eutectic composition (13.2 percent [uranium]) and containing less than 20 percent ^{235}U [uranium-235] meets the requirements of the draft standards. Additional criticality analysis of the Melt-Dilute form and HLW [high-level waste] degraded within a waste package are needed for the disposability assessment and are being scheduled for FY00 and subsequent years as part of the development process for the full scale facility. The Melt-Dilute form is flexible in that additional dilution or the addition of neutron poisons to the Melt-Dilute product can be readily made, if necessary.
>
> The Direct form in disposable canisters can meet all requirements of the Draft Standards. Criticality analyses have identified that neutron poison additions are needed to preclude criticality of degraded Al-SNF [aluminum based spent nuclear fuel] within a canister and of degraded Al-SNF and HLW within a waste package. A method is needed to incorporate neutron poisons into the canisters in the demonstration that reactivity of all possible configurations is within the acceptable limit. Several poison materials have been suggested and are being evaluated and tested for compatibility with the Al-SNF. These activities will continue throughout the development process for the full scale melt and dilute facility.

Based on the preliminary criteria and the conclusions in the Disposability Assessment, preliminary judgments can be made regarding the acceptability for disposal of the final waste forms produced under the other technologies evaluated in this EIS. Final disposal requirements will be specified by NRC; currently the final waste form produced under the Conventional Processing technology (borosilicate glass) is the best demonstrated available technology for treatment of high-level waste (55 FR 22520). Therefore, DOE has high confidence that this waste form would be acceptable for disposal in a geologic repository. The final waste form produced under the Vitrification technologies and Electrometallurgical Treatment technologies is similar to that produced under the Conventional Processing technology; thus, DOE also would have high confidence in the acceptability of their final products. For Vitrification technologies, criticality and nonproliferation concerns would need to be addressed by the dilution of the highly-enriched uranium to low-enriched uranium.

The solid form with low enrichment that would be the product of mechanical dilution could be acceptable for storage in a geologic repository. However, this technology would not be as effective from a nuclear nonproliferation perspective as other treatments (such as Melt and Dilute) because of the potential to separate the pressed or chopped depleted uranium and SNF.

Nuclear materials safeguards are one of the most important issues to be addressed for both onsite storage and transportation to a repository. Much of the aluminum-based SNF contains appreciable quantities of highly enriched uranium or plutonium. In addition to secure management, there are two basic methods for ensuring that these fissile materials have the proper safeguards: (1) reducing the uranium-235 enrichment or (2) making the fuel self-protecting. Reduced uranium-235 enrichment makes the fissile materials incapable of producing a nuclear explosion.

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Reenrichment would require a massive commitment of resources not available to most nations. "Self-protecting" means the radiation fields around the fuel are sufficiently high that recovery of the fissile materials would be impossible without the considerable resources of facilities such as those at SRS.

Finally, the integrity of the fuel form that is stored after treatment pending shipment to a repository must be sufficient to ensure safe interim storage and to prevent degradation of design features that may be relied upon in the repository.

TC Because the melt and dilute waste form could eventually be disposed of in a geologic repository, DOE-SR signed in August 1997 a Memorandum of Understanding with the NRC for its review of the research effort that DOE-SR is conducting. DOE-SR has provided the NRC with several technical reports on the results obtained from the research effort. Based upon its initial review, the NRC in a June 1998 letter (Knapp 1998) stated that "both the direct co-disposal and melt-dilute options would be acceptable concepts for the disposal of aluminum-based research reactor SNF in the repository." Additionally, as research efforts yield new findings, DOE is providing the information to the NRC.

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DOE would not implement a treatment technology option unless it has a high degree of confidence that the technology option would produce a final form that was compatible with what DOE believes the repository acceptance criteria will be. In order to ensure that the treatment technology DOE could select will produce a product that is likely to meet the acceptance criteria, DOE-SR is working with the NRC to obtain comments on the research and development work that DOE will perform to establish treatment technology specifications. To provide additional confidence in the suitability of new treatment technologies, DOE requested that the National Academy of Sciences (NAS) evaluate and provide recommendations regarding DOE's aluminum-based SNF disposition technical development program. Results of the NAS review are summarized in Section 2.6.1.

2.2.2 FACILITIES

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Under the alternatives considered in this EIS, the Department could need a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility. A Transfer and Storage Facility for SNF would provide remote handling and heavy lifting capability, hot cells, and space to receive SNF shipments; place the SNF in interim storage as needed; open the shipping containers; sample and analyze the fuel; crop end fittings if necessary; vacuum-dry the SNF; repackage the fuel into storage canisters; and place the repackaged fuel in dry interim storage. Section 2.3.2.1 provides information on the Transfer and Storage Facility. A Transfer, Storage, and Treatment Facility would provide the capability to implement the options of the New Processing Technology. Section 2.3.2.2 provides more information on the Transfer, Storage, and Treatment Facility.

For all technologies, DOE would continue to use the Receiving Basin for Offsite Fuel and the L-Reactor Disassembly Basin for currently stored SNF and to receive and store incoming fuel. If DOE built the Transfer and Storage Facility, newly received fuel could go to that facility, and the inventory in the wet basins would gradually be moved to new dry storage. DOE intends to discontinue wet storage by 2009 (DOE could continue to use the L-Reactor Disassembly Basin for SNF receipt and unloading if Building 105-L was modified as a Transfer, Storage, and Treatment Facility [see Section 2.3.2]).

All currently stored SNF at the SRS is located in the Receiving Basin for Offsite Fuel or the L-Reactor Disassembly Basin (generically termed "wet basins" in this EIS). DOE initially would receive and store incoming fuel either in the L-Reactor Disassembly Basin or the Receiving Basin for Offsite Fuel and begin construction of a new Transfer and Storage or Transfer, Storage, and Treatment Facility. Fuel would be transported from wet storage basins to the new facility as prescribed to prepare the material for disposi-

tion. Radiological consequences of the on-site transportation of the spent nuclear fuel, under both incident-free and accident conditions are projected in Section 4.1.1.7.

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2.2.3 NEW PACKAGING TECHNOLOGY OPTIONS

In this section DOE describes technology options (Direct Disposal/Direct Co-Disposal) that could be used to prepare aluminum-based SNF for placement in a geologic repository and a technology option (Repackaging and Prepare to Ship) that DOE could use to transfer non-aluminum-clad SNF and programmatic material to dry storage pending offsite shipment.

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The Direct Disposal/Direct Co-Disposal technology has the advantage of being one of the simplest to implement because it would not require a Treatment Facility, nor would it entail many operational activities. However, several potential technical issues associated with the repository must be resolved. The acceptability of aluminum-based, highly-enriched uranium fuel in a geologic repository is uncertain because of criticality concerns. DOE proposes to address this matter by limiting the amount of uranium permitted in a canister of fuel and by adding a neutron poison. Hydrogen could be produced from radiolysis of bound water in the aluminum metal fuel; however, DOE could minimize hydrogen production by adequate drying and venting, if necessary. The level of SNF characterization and certification requirements is uncertain. DOE expects the operational history of the fuel and some statistical analysis, combined with an evaluation of the more important chemical and physical characteristics (e.g., original fissile material loading, post irradiation burn-up and radiation levels) should be sufficient to characterize the fuel. The need for more detailed characterization information, based on regulatory requirements that will be developed in the future, could require much more costly and time-consuming analysis for each fuel.

2.2.3.1 Prepare for Direct Disposal/Direct Co-Disposal

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In the Transfer and Storage Facility, the SNF would be cropped (cropping removes the end pieces of the assembly; see Glossary), vacuum dried, and placed in a stainless-steel canister with a neutron poison. The canisters would be filled with an inert gas, welded closed, and placed in dry storage to await shipment to the geologic repository. Some of the uranium oxide and uranium silicide fuels could require cutting or other resizing to fit into the canisters. As an alternative, special packaging could be used for these oversized fuels.

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From an SRS perspective, Direct Disposal and Direct Co-Disposal are identical except for a slight difference in number of canisters produced. The analyses in this EIS would apply equally to either technology. If DOE used canisters with a diameter of about 17 inches (43 centimeters), it could co-dispose (see text box on page 2-3 on the co-disposal concept) the canisters at the repository with vitrified high-level waste prepared in DWPF (Direct Co-Disposal). Otherwise, using 24-inch (61-centimeter) diameter canisters, DOE could dispose of the fuel between waste packages of commercial SNF (Direct Disposal).

Due to the nature and form of the SNF to be managed at SRS, DOE does not expect the Direct Disposal/Direct Co-Disposal technology option would be applicable to all the aluminum-based SNF considered in this EIS. Table 2-1 presents an explanation of the SNF that DOE considers appropriate for the Direct Disposal/ Direct Co-Disposal option.

Figure 2-1 shows the Direct Disposal/Direct Co-Disposal option. Appendix A provides a more complete discussion of Direct Disposal and Direct Co-Disposal.

EC

2.2.3.2 Repackage and Prepare to Ship to Other DOE Sites

This technology option would apply to two specific fuel groups, and this is the only option considered for these fuel groups.

- DOE has designated management responsibilities for the stainless-steel and zirconium-clad fuels (Group F) to the Idaho National Engineering and Environmental Laboratory (60 FR28680). DOE analyzed the environmental impacts of shipping these non-aluminum-clad fuels to the Idaho National Engineering and Environmental Laboratory in the Programmatic SNF EIS (DOE 1995b).

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- The Higher Actinide Targets would be stored pending an evaluation of their disposition. Under the Repackaging and Prepare to Ship to Other DOE Sites technology option, DOE evaluates repackaging the Mark-51 and other targets to place them in a new dry storage facility in the event disposition decisions have not been made by the time an SRS dry storage facility is operational.

  DOE would not apply the Repackaging and Prepare to Ship option to the Mark-18 targets due to potential health and safety vulnerabilities as described in Section 1.5 of this EIS.

  In the Transfer and Storage Facility, the SNF and the Mark-51 and other targets could be cropped, vacuum dried, and placed in stainless-steel canisters, possibly with a neutron poison. The canisters would be filled with an inert gas, welded closed, and placed in dry storage to await shipment offsite. Figure 2-2 shows the Repackage and Prepare to Ship option which would be implemented only in parallel with an alternative that required the construction of a Transfer and Storage Facility or Transfer, Storage, and Treatment Facility. A new facility would not be constructed solely to repackage non-aluminum-based fuels and the higher actinide targets.

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2.2.4 NEW PROCESSING TECHNOLOGY OPTIONS

The New Processing Technology options would reduce the uncertainty associated with placing aluminum SNF in a geologic repository because criticality concerns would be reduced through the opportunity to adjust enrichment, add neutron absorbers, and better control geometry.

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Under these technology options, DOE initially would receive and store incoming fuel either in the L-Reactor Disassembly Basin or the Receiving Basin for Offsite Fuel. DOE would construct and operate a Transfer, Storage, and Treatment Facility (Section 2.3.2.2) to receive later shipments, and would begin to transfer the fuel inventories in the existing storage pools to this facility. DOE could use the dry storage capacity of the facility to store SNF awaiting processing and to store the processed fuel form in a road-ready condition awaiting shipment to the geologic repository.

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If a new facility was built, DOE would phaseout operation of the L-Reactor Disassembly Basin and the Receiving Basin for Offsite Fuel by 2009. In the event that Building 105-L was modified to function as the Transfer, Storage, and Treatment Facility, SNF would continue to be received and unloaded in the L-Reactor Disassembly Basin, but long-term SNF storage in the basin and in the Receiving Basin for OffsiteFuel would be phased out. The Transfer, Storage, and Treatment Facility could be located in a new or existing facility in one of the reactor areas or in a new facility in F or H Area.

Each technology option that DOE could use in the Transfer, Storage, and Treatment Facility, except Electrometallurgical Treatment, would result in an SNF form that DOE would store in road-ready condition. The use of 17-inch (43-centimeter) diameter canisters would support the co-disposal concept; however, DOE could use other canister sizes. DOE assumed a 17-inch canister for purposes of estimating costs of each technology (see Section 2.6.5). The analyses in

this EIS would apply equally to other canister sizes.

Table 2-1. Applicability commentary of the New Packaging Technology options.

	Fuel group	Prepare for Direct Disposal/Direct Co-Disposal	Repackage and Prepare to Ship to Other DOE Sites
	A. Uranium and Thorium Metal Fuels	**Applies** - These reactive metal fuels would require rigorous drying (hot vacuum drying) to ensure dehydriding and passivation of uranium metal for both short-term and repository storage.	**Does not apply** - The Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE 1995b) determined that DOE would manage aluminum SNF at SRS. DOE would not ship aluminum-based SNF to another site for storage.
	B. Materials Test Reactor-Like Fuels	**Applies** - The fissile mass loading of the canisters would be limited because of criticality concerns. DOE and NRC[a] are discussing packaging restrictions which would eliminate the possibility of criticality.	**Does not apply** - The Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE 1995b) determined that DOE would manage aluminum SNF at SRS. DOE would not ship aluminum-clad SNF to another site for storage.
	C. HEU/LEU[b] Oxides and Silicides Requiring Resizing	**Applies** - These fuels would not fit into the 17-inch (43-centimeter) diameter canister without resizing or special packaging. The highly enriched fuels present criticality concerns. The fissile mass loading of the canisters would be limited.	**Does not apply** - The Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE 1995b) determined that DOE would manage aluminum SNF at SRS.
EC	D. Loose Uranium Oxide in Cans	**Does not apply** - Group D fuels are granular and might contain particulates. Current understanding of acceptance criteria for the geologic repository would rule out acceptance of particulate fuels.	**Does not apply** - The Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE 1995b) determined that DOE would manage aluminum SNF at SRS and would ship non-aluminum fuel to INEEL.
EC	E. Higher Actinide Targets	**Does not apply** - This fuel group will be continually wet stored until DOE decides on their final disposition.	**Applies** - In the future, DOE might decide to ship these targets to another DOE site. Application of this technology to Group E fuels would include only the preparation for shipment, not the shipment itself.
	F. Non-Aluminum-Clad Fuels	**Does not apply** - The Record of Decision for the Programmatic SNF EIS designated INEEL[c] as the location for management of non-aluminum-clad SNF. SRS activities for Group F fuels are to prepare it for shipment to INEEL.	**Applies** - Under the Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE 1995b), DOE would ship non-aluminum-clad spent nuclear fuel to INEEL. DOE analyzed shipment from wet basins (DOE 1995b) which could occur under the No-Action Alternative. This technology would provide an additional action of repackaging and dry-storing Group F fuel before shipment.

a. NRC = U.S. Nuclear Regulatory Commission.
b. HEU/LEU = Highly Enriched Uranium/Low Enriched Uranium.
c. INEEL = Idaho National Engineering and Environmental Laboratory.

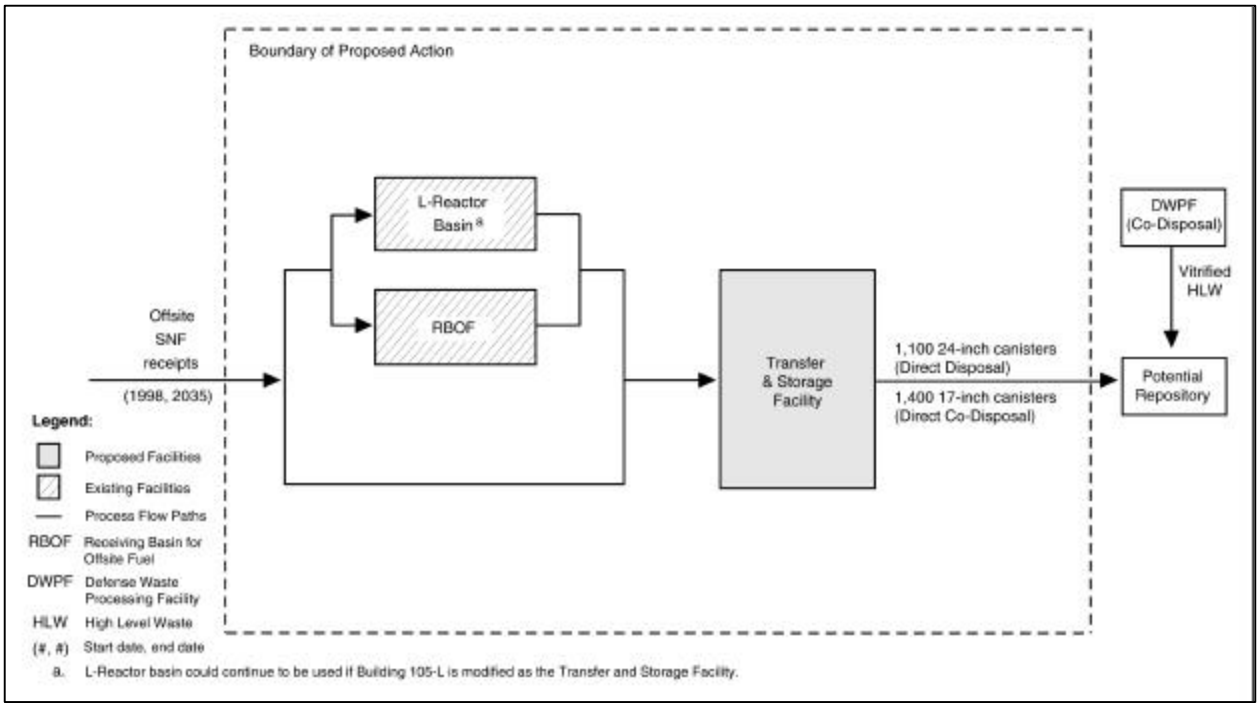

Figure 2-1. New Packaging Technology – Direct Disposal/Direct Co-Disposal.

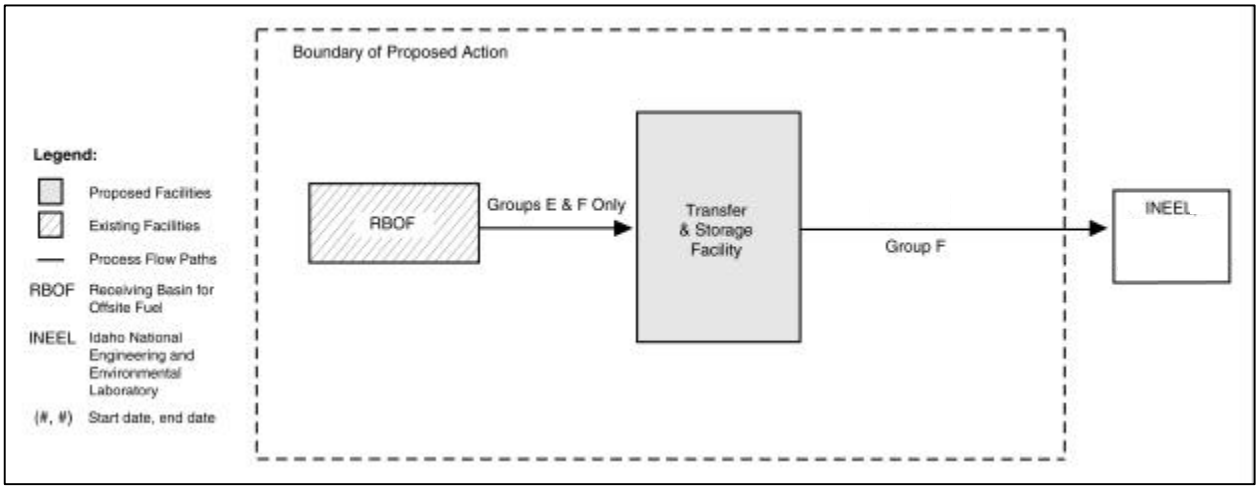

Figure 2-2. New Packaging Technology – Repackage and Prepare to Ship to Another DOE site.

Figures 2-3 and 2-4 show the New Processing Technology options. The following sections describe the new technology options; Appendix A describes them in more detail. Table 2-2 lists the applicability of the New Processing Technology to the fuel groups described in Chapter 1.

2.2.4.1 Melt and Dilute

Under the Melt and Dilute option, DOE would receive, unload, and crop the SNF in the Transfer, Storage, and Treatment Facility and either package the fuel in canisters for placement in dry storage pending treatment or send it directly to the treatment phase. The SNF would be melted and, if highly enriched, mixed with depleted ura-

nium and additional aluminum as necessary to produce a low-enriched uranium-aluminum melt. Neutron poison material also could be added if necessary. The low-enriched uranium product would be placed in corrosion-resistant canisters. The canisters, about 17-inch diameter by 120-inch length (43 by 305 centimeters), would be filled with an inert gas, welded closed, and placed in dry storage to await shipment to the geologic repository.

Under this option, most of the fission products would remain in the uranium-aluminum melt; however, some would be volatilized. Dilution to low enrichment would address nuclear proliferation concerns relating to transport and disposal of fuels. Both the dilution and the poison addition would address criticality concerns. Other characteristics promoting acceptability of the final form for disposal in the geologic repository are discussed in Appendix A.

Based on recent research and development work, preliminary conceptual design work, and considering aspects such as technical maturity, DOE considers Melt and Dilute to be the most viable of the technology options for implementation at SRS. DOE believes Melt and Dilute would entail the least technical risk because DOE has made substantial progress in the development of the melt and dilute process and ongoing work indicates full-scale operations that melt aluminum-based SNF and isotopically dilute the high-enriched uranium are achievable. A review by the National Academy of Sciences indicated that the Melt and Dilute process, as proposed by the SRS, should be achievable for aluminum-based SNF to be managed at SRS.

During the development of the Melt and Dilute technology, DOE may determine that, for technical, regulatory, or cost reasons, the Melt and Dilute option is no longer viable. As a back-up to Melt and Dilute, DOE will continue to pursue the Direct Co-Disposal option of the New Packaging Technology and would attempt to implement this option if Melt and Dilute were no longer feasible or preferable. Direct Co-Disposal has the potential to be the least complicated of the new technology options. However, there is uncertainty that aluminum-based SNF, packaged according to the Direct Co-Disposal option, would be acceptable in a geologic repository. A comparison of the preferred and backup technologies for aluminum-based nuclear fuel disposal is presented in Table 2-3.

The DOE-SR and the NRC have established an agreement for the NRC to provide technical assistance in connection with the identification of potential issues relating to the placement of aluminum-based foreign and domestic research reactor spent nuclear fuel in a geologic repository. In a review of DOE's research and development work, the NRC staff indicated that both the Melt and Dilute and Direct Co-Disposal technologies would be acceptable concepts for the disposal of aluminum- based research reactor SNF in a repository (Knapp 1998).

2.2.4.2 Mechanical Dilution

For this option, DOE would use a mechanical process to consolidate the fuel and isotopically dilute the uranium-235. The process could be either Press and Dilute or Chop and Dilute (see Appendix A). The impact analyses in Chapter 4 are based on Press and Dilute because DOE believes those impacts would be representative of both technologies, which would have nearly identical process flows, facility requirements, and resulting fuel forms.

DOE would crop and cold-vacuum-dry SNF in the Transfer, Storage, and Treatment Facility and either place the fuel in canisters for dry storage pending treatment or send the fuel directly to the treatment phase for volume reduction and dilution. The Press and Dilute method would flatten fuel assemblies and press them into a laminate between layers of depleted uranium to produce packages with a low overall enrichment. The Chop and Dilute method would shred the fuel and mix it with depleted uranium. Regardless of the dilution method, DOE would package the product in 17- by 120-inch (43- by 305-centimeter) canisters. The package could contain a nuclear poi-

son (in either the laminate or the container) to reduce the

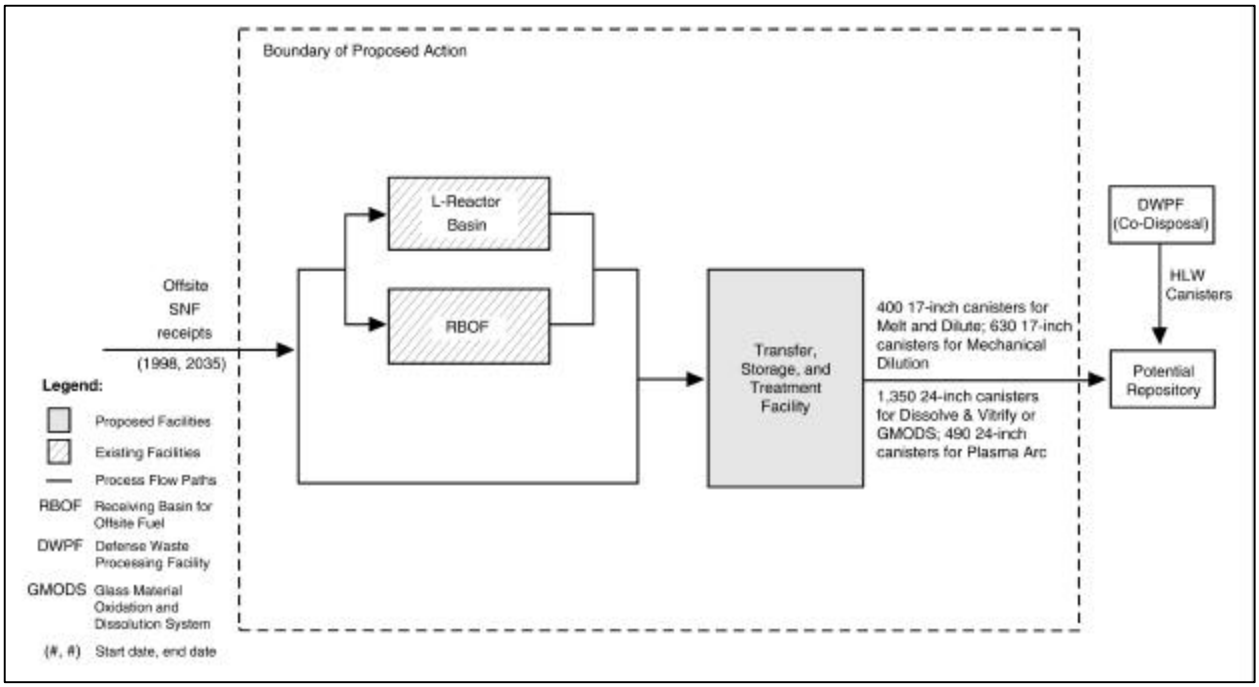

Figure 2-3. New Processing Technology - Melt and Dilute, Mechanical Dilution, Vitrification Technologies.

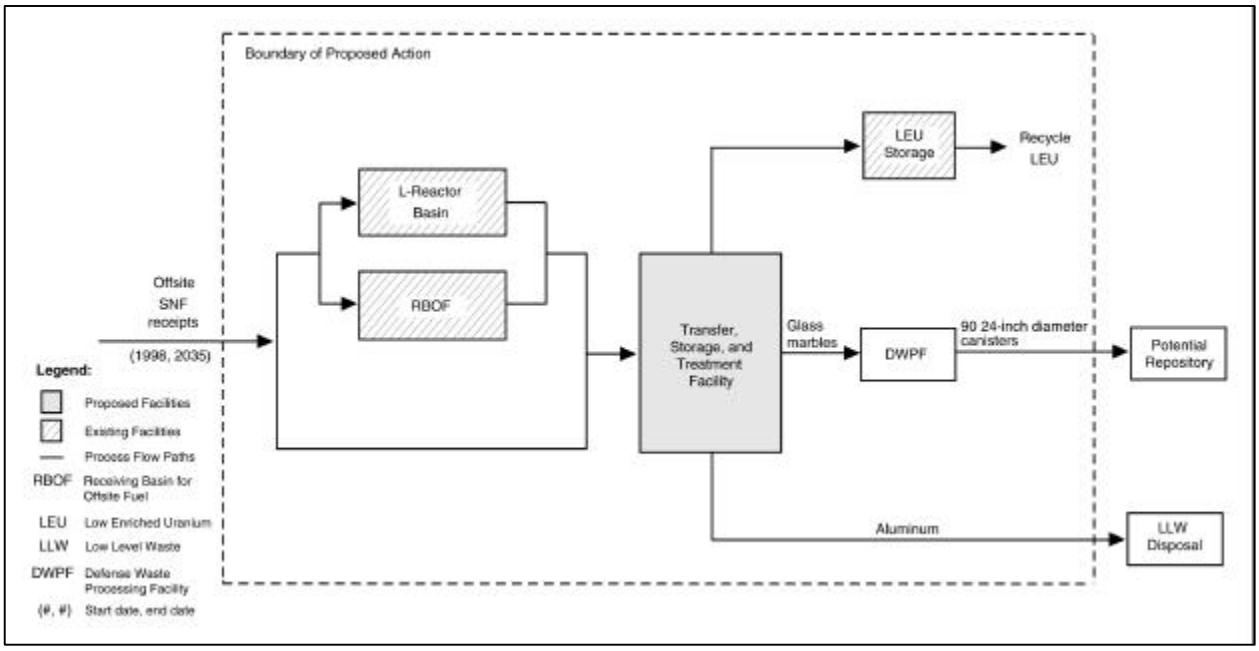

Figure 2-4. New Processing Technology – Electrometallurgical Treatment.

Table 2-2. Applicability of New Processing Technology options.

	Fuel Group	Melt and Dilute	Mechanical Dilution	Vitrification Technologies	Electrometallurgical Treatment
A.	Uranium and Thorium Metal Fuels	**Applies**	**Does not apply** - Mechanical treatment would not address chemical reactivity issue.	**Applies**	**Applies**
B.	Materials Test Reactor-Like Fuels	**Applies**	**Applies**	**Applies**	**Applies**
C.	HEU/LEU[a] Oxides and Silicides Requiring Resizing	**Applies**	**Applies**	**Applies**	**Applies**
D.	Loose Uranium Oxide in Cans	**Applies**	**Does not apply** - These fuels are granular and might contain particulates. This technology would leave Group D fuels as particulates. Current understanding of repository acceptance criteria is that particulate fuels would not be accepted without special treatment.	**Applies**	**Applies**
EC E.	Higher Actinide Targets	This fuel group will be continually wet stored until DOE decides on their final disposition.	This fuel group will be continually wet stored until DOE decides on their final disposition.	This fuel group will be continually wet stored until DOE decides on their final disposition.	This fuel group will be continually wet stored until DOE decides on their final disposition.
F.	Non-Aluminum-Clad Fuels	**Does not apply** - Record of Decision for Programmatic SNF EIS[b] designated INEEL[c] as location for non-aluminum SNF management.	**Does not apply** - Record of Decision for Programmatic SNF EIS designated INEEL as location for non-aluminum SNF management.	**Does not apply** - Record of Decision for Programmatic SNF EIS designated INEEL as location for non-aluminum SNF management.	**Does not apply** - Record of Decision for Programmatic SNF EIS designated INEEL as location for non-aluminum SNF management.

a. HEU/LEU = highly enriched uranium/low enriched uranium.
b. DOE (1995b).
c. INEEL = Idaho National Engineering and Environmental Laboratory.

Table 2-3. Comparison of preferred and backup technologies for aluminum-SNF disposal.

Technology	Advantages	Disadvantages
Preferred technology: Melt-Dilute Process	• U-235 enrichment readily adjusted by dilution with depleted uranium to meet proliferation policy and nuclear criticality constraints. • Melting reduces the volume of the fuel (see Section A.2.1). DOE estimates about 400 canisters would be generated, in comparison to about 1,400 canisters for Direct Co-Disposal. • Homogenous melt product provides basis for predictable behavior in geologic repository.	• Implementation requires high temperature operation of melter and offgas control equipment in shielded cell.
Backup technology: Direct Co-Disposal Process	• Process technically straightforward to implement. Shielded-cell handling procedures well developed. • Meets non-proliferation policy criteria better than other alternatives.	• Different SNF configurations, materials, and U-235 enrichments present packaging complexities. • No adjustment of U-235 enrichment possible to meet criticality constraints in a geologic repository. May require the use of exotic nuclear poisons. • No reduction in the volume of the fuel. • Non-uniform SNF structures and compositions complicates documentation of fuel characteristics to meet repository waste acceptance criteria and to predict behavior in a geologic repository.

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potential for criticality. The canisters would be filled with an inert gas, welded closed, and placed in dry storage to await shipment to the geologic repository.

The fission products would remain with the uranium-aluminum alloy, making their release difficult. However, mechanical dilution would not be as effective from a nuclear nonproliferation viewpoint as other treatments (such as Melt and Dilute) because of the potential to separate the pressed or chopped depleted uranium and SNF. The dilution process and the addition of a neutron poison would decrease criticality potential. The solid form with low enrichment could be acceptable at the geologic repository. Although hydrogen generation in the canister would be possible due to the radiolysis of bound water, DOE could minimize hydrogen buildup by eliminating water from the canisters (e.g., by vacuum drying).

2.2.4.3 Vitrification Technologies

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DOE could use one of three vitrification technologies: (1) Dissolve and Vitrify, (2) Glass Material Oxidation Dissolution System, or (3) Plasma Arc Treatment. In the vitrification options, the SNF would be converted to oxide and dissolved in molten glass to form a vitrified product. These options have the advantage of producing a vitrified waste form similar to that used for the disposal of high-level waste. Therefore, they should qualify for acceptance at a geologic repository. The final form would contain fission products, and criticality and nonproliferation concerns would be addressed by the dilution of enriched uranium.

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For these options, DOE would crop and cold-vacuum-dry SNF in the Transfer, Storage, and Treatment Facility and either place the fuel in canisters for dry storage pending treatment or send it immediately for treatment. The resulting glass or ceramic would be poured into 24- by

120-inch (61- by 305-centimeter) canisters and placed in dry storage. The use of 24-inch diameter canisters would enable disposal like vitrified high-level waste.

These are advanced technologies. As such, they introduce more technical and schedule risk than the other options in this alternative. This EIS analyzes the impacts of the Dissolve and Vitrify option as representative of all three because DOE believes that the impacts among the three would be similar. The following paragraphs describe the three vitrification technologies; Appendix A provides more information.

Dissolve and Vitrify

The Dissolve and Vitrify treatment is similar to conventional processing except there would be no recovery of enriched uranium. The SNF would be cropped and charged to an electrolytic dissolver. The electrolyte solution would be nitric acid saturated with boric acid. If necessary, depleted uranium would be added to produce low-enriched uranium. The entire solution, including uranium and fission products, would be vitrified. The process would operate in a batch mode to ensure criticality control.

This EIS analyzes performing the Dissolve and Vitrify option in the Transfer, Storage, and Treatment Facility; however, DOE could modify one of the canyons to perform the process. DOE is not considering vitrification of this material in DWPF because that process is not designed to accommodate more than trace quantities of fissile material without major modifications that would be impractical and incompatible with DWPF operations, schedules, and mission.

Glass Material Oxidation and Dissolution System

The Glass Material Oxidation and Dissolution System would convert SNF directly to borosilicate glass using a batch process. The final form would address criticality concerns by diluting the uranium-235 with depleted uranium and by using boron oxide as a dissolving agent (boron is a neutron poison).

The process would use lead dioxide to oxidize the metals in the SNF so they would be soluble in glass. The resulting lead metal would be recovered and oxidized for reuse. The product of the process would be glass marbles that a second stage of melting could consolidate into logs. The process would occur in the new Transfer, Storage, and Treatment Facility.

Plasma Arc Treatment

The Plasma Arc Treatment technology would use a plasma torch to melt and oxidize the SNF in a rotating furnace. The fuel would be fed into the process with minimal sizing or pretreatment. The plasma torch would heat the fuel to temperatures as high as 2,900°F (1,600°C). The rotation of the furnace and the pressure of the torch would mix the melted fuel. A ceramic binder such as contaminated soil would be added to the mixture to form a glass-ceramic. Depleted uranium could be added to the process to produce low-enriched uranium. When the melting and oxidation is complete, the furnace rotation would slow and the molten fuel would flow by gravity into molds. The process would be conducted in the Transfer, Storage, and Treatment Facility, which would be equipped to capture volatile and semivolatile off-gasses.

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2.2.4.4 Electrometallurgical Treatment

Under the Electrometallurgical Treatment option, DOE would crop and cold-vacuum-dry the SNF in the Transfer, Storage, and Treatment Facility, can it, and either place it in dry storage pending treatment or send it immediately to the treatment phase, which would shred and melt it into metal ingots. An ingot would be placed in an electrorefiner, where most of the metal in the SNF (aluminum) would be removed as a low-level waste stream. The remaining metal would be placed in a second electrorefiner where the uranium would be removed. If necessary, the uranium would be fed to a melter where depleted uranium would be added to produce low-enriched uranium. The

uranium could be sold as recycled uranium for manufacture into commercial nuclear fuel. The remainder of the fuel materials would be oxidized in a furnace and dissolved in glass which would be poured into 24- by 120-inch (61- by 305-centimeter) canisters and placed into dry storage.

This option has the advantage of potentially recycling the enriched uranium. Criticality concerns would be addressed by the isotope dilution of the highly enriched uranium, eliminating the issue of SNF acceptance at a geologic repository. DOE has been developing the electrometallurgical treatment process for certain non-aluminum-based SNF.

Figure 2-4 shows the Electrometallurgical Treatment technology. Appendix A provides a more complete discussion of the technology.

2.2.5 CONVENTIONAL PROCESSING TECHNOLOGY

In this technology, DOE would process SNF in the F or H Area Canyon directly from wet storage. The Record of Decision for the Final EIS on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (61 FR 25091) stated that fuel would be processed in F Canyon. Because F Canyon is scheduled to be shut down before all the fuel could be processed, and because F Canyon is not suitable for highly-enriched uranium processing without modifications, H Canyon also would be used. The process would chemically dissolve the fuel and separate fission products from the uranium by solvent extraction. The uranium would be blended with depleted uranium, as necessary, to bring the enrichment down to about 5 percent or less. The wastes from solvent extraction would contain the highly radioactive fission products, thorium, and possibly some uranium. This high-level waste would be separated into high- and low-activity fractions, which would be converted to glass (vitrified) in DWPF and to a cementitious low-level solid in the Saltstone Manufacturing and Disposal Facility, respectively. Recovered uranium could be sold to a commercial producer of nuclear fuel. DOE would dispose of the vitrified waste in a geologic repository and the saltstone in onsite vaults.

For Conventional Processing, DOE would use several existing SRS facilities:

- The L-Reactor Disassembly Basin and the Receiving Basin for Offsite Fuel for interim storage of the SNF before processing
- The F and H Canyons and related facilities for processing
- The high-level waste tank farms, DWPF, and Saltstone Manufacturing and Disposal Facility for high-level waste disposition

DOE expects that the Experimental Breeder Reactor-II fuel and the Mark-42 targets would be processed in F Canyon. The operation would result in the separation of plutonium that would be converted to metal in FB-Line and then placed in storage at SRS pending disposition in accordance with decisions reached under the *Surplus Plutonium Storage and Disposition EIS* currently being prepared by DOE. This material would not be used in any military application. All other processing operations would be conducted in H Canyon. Processing operations in H Canyon would continue if all fuel were to be processed until the aluminum-based SNF inventory was eliminated and the SNF receipt rate was low in about 2009 (i.e., receipts would be about 150 Materials Test Reactor-like elements per year and 12 High Flux Isotope Reactor assemblies per year). In parallel with processing operations, DOE could construct a Transfer, Storage, and Treatment Facility to receive and treat new SNF after processing operations cease. Because of the small volume of SNF to be processed in this facility, its dry storage capacity would be much less than required for other technologies.

Conventional Processing would be applicable to all fuel groups except most of the higher actinide targets (specifically the Mark-51 and "other" targets) and the non-aluminum-clad fuels. Con-

ventional Processing would apply to the Mark-18s in the Higher Actinide Targets fuel group. The Record of Decision for the Programmatic SNF EIS (DOE 1995b) designated the Idaho National Engineering and Environmental Laboratory as the location for management of non-aluminum-clad SNF. The SRS would store these fuels pending shipment to the Idaho National Engineering and Environmental Laboratory.

The resulting low-enriched uranium would not be suitable for use in weapons and any plutonium separated from the Experimental Breeder Reactor-II fuel or Mark-42 targets would be part of the plutonium considered surplus to the nuclear weapons program that will be dispositioned through decisions reached under the plutonium disposition EIS. Repository acceptance criteria should not be an issue because the vitrified high-level waste would be the same as the vitrified waste DOE is currently producing at SRS, and DOE has a high level of confidence that vitrified waste will meet the repository acceptance criteria. This option would add to the inventory of waste stored at SRS. However, sufficient storage and DWPF capacity exist to accommodate the added volume.

Figure 2-5 shows the Conventional Processing option. Appendix A provides more information on the technology.

2.3 Spent Nuclear Fuel Management Facilities

The implementation of the proposed action would require the construction of a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility and the use of several existing facilities, depending on the alternative selected. Table 2-4 lists the facilities required for the technologies. The following sections describe the existing and new facilities.

2.3.1 EXISTING FACILITIES

The existing SRS facilities that DOE would need for the proposed action are the L-Reactor Facility, the Receiving Basin for Offsite Fuel, and the F and H Canyons. Figure 2-6 shows the locations of these facilities. Appendix B provides information on the status of identified vulnerabilities at these facilities.

2.3.1.1 L-Reactor Facility

Facility Description

The Federal Government built L Reactor in the early 1950s to produce nuclear materials for national defense. In 1988 DOE shut the reactor down for safety upgrades, and has not restarted it. In 1993 the Department ended the reactor's materials production mission. The current mission of this facility is to store reactor components and other radioactive materials in the disassembly basin, receive and store foreign and domestic research reactor fuel in the disassembly basin, decontaminate shipping casks in the stack area, store contaminated moderator in tanks or drums, and compact low-level waste in a compactor. DOE maintains the structures, systems, and components necessary to perform these missions, but has deenergized, drained, or otherwise deactivated many others.

In addition to the support systems, L Reactor has three principal areas that could be important to the proposed action – the disassembly basin, the L-Reactor building, and the stack area. Figure 2-7 shows L-Reactor and indicates the locations of these areas.

The disassembly basin, which would be the principal structure supporting the SNF storage mission, is a large concrete basin containing approximately 3.4 million gallons (13,000 cubic meters) of water varying in depth from 17 to 50 feet (5.2 to 15 meters). DOE has upgraded the basin to improve water control and monitoring, including continuously operating deionizers to improve water chemistry, makeup water deionizers, and a water level monitoring system. In addition, DOE has added storage racks to accommodate anticipated fuel receipts. The disassembly basin contains a transfer bay with one water-filled pit and heavy lifting equipment to transfer shipping casks to the basin.

The L-Reactor building has space potentially suitable for installation of facilities for treatment of SNF (see Section 2.3.2.2). The space includes the process room and crane maintenance area. The process room, a shielded area situated

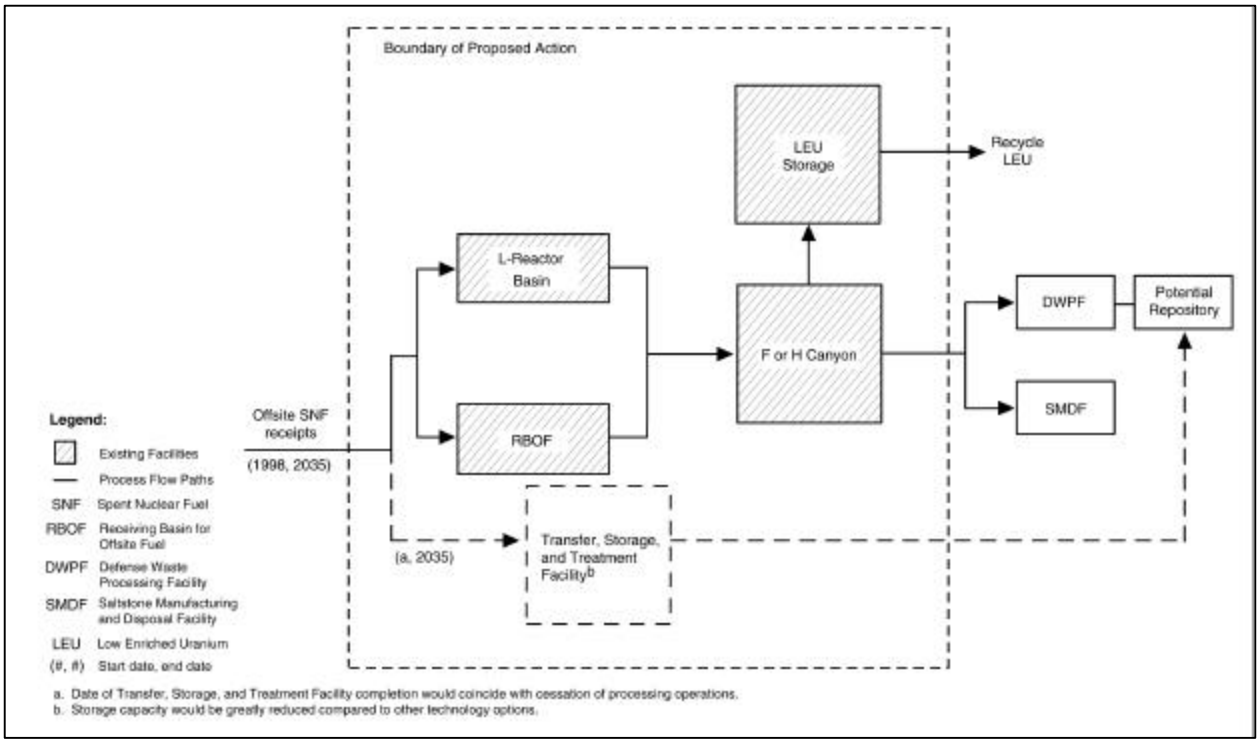

Figure 2-5. Conventional Processing.

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Table 2-4. Facilities needed for SNF technologies.

Technology	Receiving Basin for Offsite Fuel	L-Reactor Facility	F or H Canyon	Transfer and Storage Facility	Melt and Dilute Treatment Facility	Mechanical Dilution Treatment Facility	Vitrification Facility	Electrometallurgical Treatment Facility	Renovated Reactor Facility
1. Prepare for Direct Disposal/Direct Co-Disposal	✓	✓		✓					✓
2. Repackage and Prepare to Ship[a]	✓	✓[b]		✓					✓
3. Melt and Dilute	✓	✓		✓	✓				✓
4. Mechanical Dilution	✓	✓		✓		✓			✓
5. Vitrification Technologies	✓	✓		✓			✓		✓
6. Electrometallurgical Treatment	✓	✓		✓				✓	✓
7. Conventional Processing	✓	✓	✓	✓	✓[c]				
8. Continued Wet Storage	✓	✓							

a. To another DOE site.
b. Needed only if a Transfer, Storage, and Treatment Facility were implemented in a reactor facility.
c. Once conventional processing is terminated, the remaining SNF would require treatment using one of the new technologies. A Melt and Dilute Treatment Facility is included as part of Conventional Processing as a reference follow-on treatment

above the reactor tank, formerly provided access to the reactor by means of a charge and discharge machine for handling reactor fuel assemblies. The area is serviced by an overhead crane. Fuel assemblies were transferred from the L-Reactor Disassembly Basin to the process room by way of an interconnecting water canal. The crane maintenance area, connected to the process room by a shielded crane wash area, allowed hands-on maintenance of the fuel assembly transfer systems.

DOE uses the L-Reactor stack area to unload shipping casks from their International Organization for Standardization (ISO) containers and to decontaminate empty shipping casks. The decontamination hut has a sump pump, spray equipment, a ventilation system, and deionizers.

In 1993 DOE performed a vulnerability assessment of its SNF facilities and identified several vulnerabilities related to the disassembly basins (DOE 1993). The Defense Nuclear Facilities Safety Board reported other vulnerabilities (DNFSB 1994; Burnfield 1995; Conway 1996), including the lack of adequate water chemistry control, which resulted in the corrosion of stored SNF and some cladding failure. The corroding fuel resulted in a buildup of radionuclides in the water and in the sludge at the bottom of the basins. Another vulnerability was the lack of an adequate leak detection capability. Since the vulnerability assessments, DOE has completed the corrective actions. One of the more significant upgrades is the installation of deionizers for maintaining water quality; maintenance of water chemistry is important to minimize corrosion. Appendix B describes these vulnerabilities and corrective action plans in greater detail.

Facility Operations

EC DOE would receive SNF in shipping casks designed to meet SNF cask design criteria (10 CFR 71). If the cask was too large for the L-Reactor Disassembly Basin or if other operational restrictions (such as a maintenance out-age) occurred, DOE would transport the cask to the Receiving Basin for Offsite Fuel in H Area, remove the fuel and place it in a smaller cask, and transfer it to L Reactor. The smaller casks would be moved to the transfer bay of the disassembly basin.

SNF is unloaded from the casks underwater. The procedure is as follows: the casks are vented, filled with water, and submerged in the transfer bay. The purged air is cleaned by high-efficiency particulate air filters before being discharged to the atmosphere. The casks are opened and the fuel elements placed in a bucket for examination. If the fuel cannot be identified or is inconsistent with the documentation provided by the reactor operator, it is isolated until the discrepancy is resolved.

The SNF is moved to the storage area of the disassembly basin through a transfer canal. The cask lid is replaced and the cask is drained, washed, and decontaminated. Decontamination water is sent to the disassembly basin.

2.3.1.2 Receiving Basin for Offsite Fuel

Facility Description

The Receiving Basin for Offsite Fuel, located in H Area, has provided storage for irradiated SNF since 1964. It has an unloading basin, two storage basins, a repackaging basin, a disassembly basin, and an inspection basin, all underwater. Fuel is handled or stored under at least 4 feet (1.2 meters) of water to provide shielding against radiation. The reinforced-concrete basins are below grade. They have either chemical coatings or stainless-steel linings for ease of decontamination. The storage lattice in the basins consist of rows of racks of aluminum I-beams. Gratings, guide plates, and spacers between the racks separate individual storage positions and provide the spacing required for criticality safety. EC

In addition to the water-filled basins, the Receiving Basin for Offsite Fuel has a receiving bay, dry cask inspection pit, control room, office areas, equipment storage areas, and concrete cells that contain tanks for water decontamination (deionization) and temporary storage of

Figure 2-6. SRS map indicating locations of facilities needed for Proposed Action.

Figure 2-7. Plan view of the L-Reactor facility.

radioactive liquid waste. The facility has a 100-ton (91-metric-ton) bridge crane that travels on rails approximately 31 feet (9 meters) above grade. The crane has two 50-ton (45-metric-ton) hoists and two 3-ton (2.7-metric-ton) hoists. The crane travels over the cask receiving, unloading, and fuel storage areas.

The DOE vulnerability assessment (DOE 1993) and inspections performed by the Defense Nuclear Facilities Safety Board (Burnfield 1995; Conway 1996) identified vulnerabilities related to the Receiving Basin for Offsite Fuel. These vulnerabilities primarily involved the seismic qualification of the building, the lack of adequate leak detection, and the spacing of vertically stored fuel assemblies (a criticality concern). Appendix B describes these vulnerabilities and their corrective actions (which have all been completed).

Facility Operations

The receiving bay on the north side of the Receiving Basin for Offsite Fuel receives shipping casks containing irradiated fuel delivered by truck or rail. Radiological surveys of the casks determine external radiation and surface contamination levels. The cask is vented after cleaning and filled with water that is sampled to detect contamination, which would indicate damaged or failed fuel. The cask lid bolts are loosened and the cask transferred to the cask basin using the 100-ton (91-metric-ton) overhead crane. The cask is lowered into the basin until the top of the lid is approximately 3 feet (1 meter) above the water surface and the lid bolts are removed. The cask is lowered to the bottom of the basin and the lid removed. Fuel elements are removed from the cask and placed in transfer buckets, cans, or bundles, depending on the fuel design. The bucket, can, or bundle is placed in a storage rack and the process repeated until all fuel had been unloaded from the cask.

The Receiving Basin for Offsite Fuel has separate basins to segregate and can damaged or failed fuel, disassemble fuel components by mechanical means (e.g., cutting), or perform inspection and measurement. The basin water circulates through a filter and a deionizer for purification and clarification. DOE replaces the filters and deionizers periodically, depending on radioactivity or impurity levels in the water.

2.3.1.3 F and H Canyons

Facility Description

Two SRS facilities – F and H Canyons – could chemically separate uranium from fission products in SNF. The canyon facilities are nearly identical and use similar radiochemical processes for the separation and recovery of plutonium, neptunium, and uranium isotopes. Historically, F Canyon recovered plutonium-239 and uranium-238 from irradiated natural or depleted uranium, and H Canyon recovered pluto-nium-238, neptunium-237, and uranium-235 from irradiated reactor fuels and targets.

The canyons buildings are reinforced-concrete structures, 835 feet (254 meters) long by 122 feet (37 meters) wide by 66 feet (20 meters) high. They house the large equipment (tanks, process vessels, evaporators, etc.) used in the chemical separations processes.

Each canyon facility contains two canyons, the hot canyon and the warm canyon. The two canyons are parallel and separated by a center section, which has four floors. The center section contains office space, the control room for facility operations, chemical feed systems, and support equipment such as ventilation fans. Processing operations involving high radiation levels (dissolution, fission product separation, and high-level radioactive waste evaporation) occur in the hot canyon, which has thick concrete walls to shield people outside and in the center section from radiation. The final steps of the chemical separations process, which generally involve lower radiation levels, occur in the warm canyon. The F and H Canyons are designed to prevent the release of airborne radioactivity. The ventilation systems maintain a negative air pressure with respect to outside pressure. The ventilation discharges are filtered by high-efficiency particulate air filters and sand filters that remove

EC

more than 99.9 percent of the particulate radioactivity. Figure 2-8 shows a cutaway view of a canyon building. Figure 2-9 is an aerial photograph of H Canyon and the surrounding area.

DOE and the Defense Nuclear Facilities Safety Board have identified environmental, safety, and health vulnerabilities at the F and H Canyons (DOE 1993; DNFSB 1994). These vulnerabilities relate to the seismic qualification of the buildings and the continued storage of in-process nuclear materials. DOE has verified the seismic qualification of the canyons. In accor-dance with the various Records of Decision for the Interim Management of Nuclear Materials EIS (DOE 1995a), DOE is stabilizing selected materials of concern identified by the Defense Nuclear Facilities Safety Board.

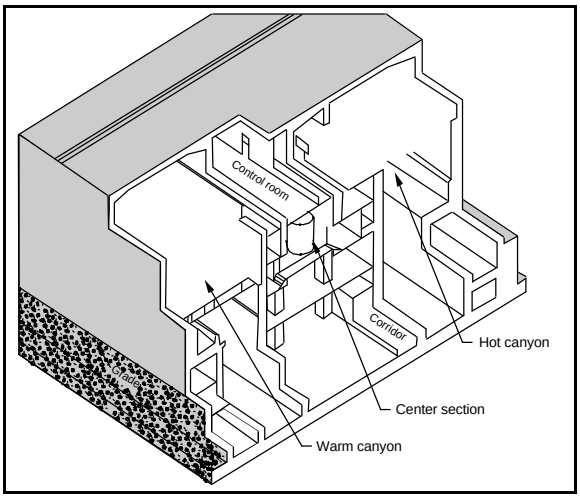


Figure 2-8. Canyon building sections.

Facility Operations

The SNF would arrive by rail in a shielded shipping cask from either the Receiving Basin for Offsite Fuel or the L-Reactor Disassembly Basin. The fuel would be unloaded and placed in an interim storage pool by a remotely operated crane. At the appropriate time, the fuel would be placed in the dissolver and dissolved by nitric acid. If the processing was performed in F Canyon, the acid solution would be blended down with depleted uranium. However, because H Canyon is designed to handle enriched uranium, the blending to low enriched uranium in H Canyon could occur at virtually any point in the processing operation. In either case, the uranium would be blended to about 5 percent uranium-235.

The resulting acid solution would be chemically processed using clarification and solvent extraction to produce a relatively pure and concentrated stream of uranyl nitrate, which would be stored in tanks awaiting disposition including selling it to commercial reactor fuel users/ manufacturers. Building ventilation discharge would be filtered (including sand filters) to remove at least 99.9 percent of the particulate radioactivity.

2.3.2 Proposed Facilities

DOE could construct new facilities or modify existing ones to accomplish the Proposed Action, depending on the alternative selected.

2.3.2.1 Transfer and Storage Facility

A Transfer and Storage Facility would provide remote handling and heavy lifting capability, hot cells, and space to receive SNF shipments. This facility would place SNF in interim storage as needed, open the shipping containers, sample and analyze the fuel, crop end fittings if necessary, vacuum-dry the SNF, repackage the fuel in storage canisters, and place the repackaged fuel in interim storage. DOE would use this facility to perform the functions listed in Table 2-5 without the use of water-filled storage pools; however, DOE could choose to provide the capability to receive incoming SNF in a wet basin. This small wet basin, if used, would be for receipt only - not storage. Figure 2-10 shows this facility.

The dry storage segment of the facility would provide lag storage for SNF waiting for preconditioning or treatment, road-ready storage for fuel packaged for shipment to a geologic repository, and temporary storage for empty canisters and loaded and unloaded transportation casks. The size of the storage facility would depend on how DOE decided to implement the Proposed Action. For example, if DOE TC

Figure 2-9. H Canyon and surrounding area (view toward northeast).

Table 2-5. Transfer and Storage Facility functions.

Function	Description
Receiving/shipping	Receive casks, unload SNF, load casks, and prepare loaded and unloaded casks for shipment
Characterization	Inspect SNF for storage, conditioning, and disposition (e.g., visual inspection, gamma spectrometry, and calorimetry)
Conditioning	Crop end fittings or binding pins; activity would not breach cladding or modify the fuel matrix
Packaging	Place SNF in appropriate cans and canisters (e.g., vacuum drying, filling with inert gas) and packaging for road-ready storage or direct transport
Stability/verification testing	Provide analytical capabilities to perform sampling and analysis to verify conformance to repository waste acceptance criteria
Treatment Facility Interface	Provide interfaces necessary to accommodate various treatment technologies
Storage	Provide dry road-ready storage using modular design and construction

Figure 2-10. Schematic cut-away of the transfer storage and treatment facility.

selected Electrometallurgical Treatment as a new processing technology, the storage component of the facility would only need to provide lag storage for fuel awaiting treatment; no road-ready storage would be necessary because waste produced from the Electrometallurgical Treatment would be sent to DWPF. Table 2-6 lists the number of road-ready canisters DOE would need to store for each technology. In each case, the number of canisters for the treatment technologies is less than that for the Direct Co-Disposal technology. The size of the transfer operations component of the facility would be independent of any new technology selected. In the event Conventional Processing is implemented, the size of the Transfer and Storage Facility would be reduced by about 30 to 60 percent.

The storage segment probably would have one of the three generic designs shown in Figure 2-11. Regarding the environmental impacts of constructing and operating a dry storage facility, the *Foreign Research Reactor Spent Nuclear Fuel EIS* (DOE 1996c) concluded, "There are significant differences between these technologies in terms of construction, operations and maintenance costs and various design details. However, these differences do not result in any important variations in environmental impacts and consequences."

The modular dry storage vault design is a self-contained concrete structure that would provide storage for hundreds of SNF assemblies. The vault would contain a charge and discharge bay with an SNF-handling machine above a floor containing steel tubes to house the removable fuel canisters. The bay would be shielded from the stored fuel by the thick concrete floor and shield plugs inserted at the top of the steel storage tubes. Large labyrinth air supply ducts and discharge chimneys would permit natural convection cooling of the fuel storage tubes to dissipate decay heat. The perimeter concrete walls would provide shielding.

Table 2-6. Road-ready storage capacities.

Technology	Number of co-disposal canisters (17-inch diameter)
Prepare for Direct Co-Disposal/Direct Disposal	1,400[a]
Repackage and Prepare to Ship	0
Melt and Dilute	400
Mechanical Dilution	630
Vitrification Technologies	1,350[b]
Electrometallurgical Treatment	0[c]
Conventional Processing	0[d]
Continued Wet Storage	0

a. Direct Disposal in 24-inch diameter canisters would require 1,100 canisters.
b. Vitrification Technologies would produce 24-inch diameter canisters. The value reported is for Dissolve and Vitrify and Glass Material Oxidation and Dissolution System. Plasma Arc Treatment would produce 490 24-inch diameter canisters.
c. Electrometallurgical treatment would produce about 90 high-level waste canisters to be stored in the Glass Waste Storage Building of the Defense Waste Processing Facility.
d. Conventional Processing would result in storage of about 150 high-level waste canisters in the Glass Waste Storage Building of the Defense Waste Processing Facility.

Figure 2-11. Typical spent nuclear fuel dry storage facilities.

A dry concrete storage cask, either vertical cask-on-pad or a horizontal concrete module, would perform a similar function, but would not be in a vault. The cask would provide the shielding. A dedicated truck and trailer would transport the fuel containers from the transfer area of the facility to the dry storage area. A ram (for horizontal modules) or a crane (for vertical modules) would insert the fuel package into the storage cask. Appendix F of the *Foreign Research Reactor Spent Nuclear Fuel EIS* (DOE 1996c) contains more information on dry storage facility designs.

DOE used a formal site selection process (Wike et al. 1996) to identify and evaluate potential sites for the construction of the Transfer and Storage Facility. Among the siting criteria were engineering and operational parameters; infrastructure support; human health, environmental, and ecological impacts; regulatory criteria; and land use planning. The process identified five potential sites, two of which received substantially higher scores than the others. These sites are the east side of L Area inside the facility fence, and the southeast side of C Area inside the facility fence. DOE has determined that these two sites are preferred and has completed some geotechnical evaluations on them. Figures 2-7 and 2-12, respectively, show these locations. DOE has considered these two sites in the analyses in this EIS. The transfer functions performed by a Transfer and Storage Facility could also be located in a renovated reactor building. Storage facilities would be as described above.

2.3.2.2 Transfer, Storage, and Treatment Facility

DOE could build a new Transfer, Storage, and Treatment Facility in the locations previously described for the Transfer and Storage Facility. Alternatively, the facility could be located in a new facility in F or H Area (Figures 2-13 and 2-14) to take advantage of existing services and infrastructure in these areas. DOE would construct this facility only if it selected a technology that required it. The facility would be similar to the Transfer and Storage Facility described in Section 2.3.2.1, but with the addition of SNF treatment capability as described in the following paragraphs. The operations performed in the facility would depend on the treatment technology DOE selected, and could include Melt and Dilute, Mechanical Dilution, Vitrification Technologies, or Electrometallurgical Treatment.

The facility design would address criticality issues during normal operations and under conditions of extreme natural phenomena. The facility would contain hot cells, remote handling equipment for the fuel and canisters, processing equipment such as melters (depending on the technology option selected), waste handling and treatment capability, canister decontamination capability, and infrastructure needed for radiological protection operations (e.g., monitoring equipment and protective clothing change rooms). Treatment and handling operations would be performed in facility areas especially designed to prevent the release of airborne radioactivity. For example, the ventilation system would maintain a negative air pressure with respect to outside pressure. The ventilation discharge would be filtered to remove at least 99.9 percent of the particulate radioactivity. EC L3-7

DOE also is considering performing SNF treatments in a renovated reactor facility. In this EIS, DOE has evaluated modifying Building 105-L, and DOE considers this evaluation representative of other reactor area facilities. The processes for transfer and treatment would be located within the L-Reactor building (Figure 2-7), supported by capabilities in the existing structure and adjacent L-Area enclosure. The treatment facilities would be operated in close conjunction with the underwater storage of the SNF in the L-Reactor Disassembly Basin, converting the SNF to the final waste form for dry storage in a Storage Facility as described in Section 2.3.2.1.

Figure 2-12. Plan view of C-Reactor facility.

Figure 2-13. Potential Transfer, Storage, and Treatment Facility location in F Area.

Figure 2-14. Potential Transfer, Storage, and Treatment Facility location in H Area.

Table 2-7. Fuel groups and technology options that could be applied to meet the purpose and need. For each fuel group, the technologies that would produce the lowest and highest impacts have been identified.

	Fuel group	1. Prepare for Direct Co-Disposal	2. Repackage and Prepare to Ship[a]	3. Melt and Di-lute	4. Mechanical Dilution	5. Vitrification Technologies	6. Electro-metallurgical Treatment	7. Conventional Processing
A.	Uranium and Thorium Metal Fuels	Yes[b], LB[c]	No	Yes	No	Yes	Yes	Yes, UB[d]
B.	Materials Test Reactor-Like Fuels	Yes, LB	No	Yes	Yes	Yes	Yes	Yes, UB
C.	HEU/LEU[e] Oxides and Silicides Requiring Resizing or Special Packaging	Yes, LB	No	Yes	Yes	Yes	Yes	Yes, UB
D.	Loose Uranium Oxide in Cans	No	No	Yes, LB	No	Yes	Yes	Yes, UB
E.	Higher Actinide Targets[f]	NA	Yes, LB/UB	NA	NA	NA	NA	NA
F.	Non-Aluminum Clad Fuels[f]	NA	Yes, LB/UB	NA	NA	NA	NA	NA

a. This alternative describes repackaging for storage at SRS pending shipment offsite.
b. "Yes" indicates that the technology can be applied to the fuel group. "No" indicates that the technology cannot be applied to the fuel group.
c. LB = lower bound of impacts.
d. UB = upper bound of impacts.
e. HEU = highly enriched uranium; LEU = low enriched uranium.
TC f. NA = not applicable; not decided in this EIS. Higher actinide targets would be stored until DOE determined their disposition and non-aluminum clad fuel is scheduled to be shipped to Idaho National Engineering and Environmental Laboratory for treatment. Only the impacts of storing these materials are considered in this EIS.

2.4 Alternatives Evaluated

As indicated in Sections 2.2.1 through 2.2.3, none of the technologies is likely to be applicable to all the fuel groups. Table 2-7 lists the technology options DOE believes are applicable to the fuel groups discussed in this EIS. DOE probably would implement a combination of options to accomplish SNF management at SRS. Many (more than 700) technology-fuel group configurations can be created using the information in Table 2-7. Tables 2-1 and 2-2 summarize the basis for the applicability of the New Packaging options and the New Processing Technology options. Conventional Processing could be applied to any fuel group except the non-aluminum-clad fuels and the higher actinide targets. Although the No-Action Alternative could be applied to all fuel groups, it would not meet the purpose and need for action.

Taking into consideration the technology options available to the various fuel groups and decisions previously made about managing certain types of SNF, DOE developed five alternatives to analyze in this EIS. DOE has chosen to present impacts from the No Action Alternative, the Preferred Alternative, the Direct Disposal Alternative, and the Maximum- and Minimum-Impact Alternatives described below to illustrate the range of impacts that could occur from any configuration the decisionmakers might select (Table 2-8). These configurations are representative of the range of those DOE could select to accomplish the proposed action and are expected to include the upper and lower bounds of potential impacts. The No Action Alternative represents the impact from current operations.

DOE recognizes that a combination of technology options might not result in the lowest or highest impact for all evaluated technical parameters (e.g., for a particular configuration, worker health and public health impacts could be lowest, but radioactive waste generation could be highest) and that there are other reasonable alternative configurations that would result in similar minimal or substantial impacts. Impacts resulting in human health effects and environmental pollution received greater weight than those resulting in the consumption of natural resources or waste disposal space. In addition, impacts to the general public received greater weight than those to SRS workers. Similarly, impacts that would occur immediately (e.g., operation of new and existing processing facilities) received greater weight than impacts that are not expected but could occur in the distant future.

2.4.1 MINIMUM IMPACT ALTERNATIVE

This alternative consists of the fuel groups and technologies that DOE believes would result in the lowest overall impact. The identification of the minimum impact (and environmentally preferred) alternative required both quantitative and qualitative analyses. The first step tabulated the analytical parameters (e.g., volume of high-level waste, air concentrations) and the minimum-impact technology for each parameter for each fuel group. The selected analysis parameters often resulted in a combination of high and low impacts for a particular fuel group. Therefore, the second step required a qualitative examination of trends in combinations that would provide overall minimum impacts.

DOE believes that the range of impacts from other reasonable choices of the minimum-impact alternative would be small. Therefore, DOE expects that the impacts of this alternative would be representative of the lower bound of impacts from the Proposed Action.

The minimum impact alternative would include New Packaging and New Processing Technologies options. Material Test Reactor-like fuels and highly enriched uranium/low enriched uranium (HEU/LEU) oxides and silicides would be treated using the Direct Disposal/Direct Co-Disposal option and placed in the Transfer and Storage Facility with a minimum of treatment (e.g., cold-vacuum drying and canning). The uranium and thorium metal fuels would be treated using the Direct Disposal/Direct Co-Disposal option but more rigorous treatment (i.e., hot-vacuum drying) would be required.

Table 2-8. Alternatives analyzed in this EIS.

Fuel Group	No-Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative
A. Uranium and Thorium Metal Fuels	Continued Wet Storage	Prepare for Direct Co-Disposal	Conventional Processing	Conventional Processing	Conventional Processing
B. Materials Test Reactor-like Fuels	Continued Wet Storage	Prepare for Direct Co-Disposal	Prepare for Direct Co-Disposal	Melt and Dilute	Conventional Processing
C. HEU/LEU Oxide and Silicides Requiring Resizing or Special Packaging	Continued Wet Storage	Prepare for Direct Co-Disposal	Prepare for Direct[a] Co-Disposal	Melt and Dilute[a]	Conventional Processing
D. Loose Uranium Oxide in Cans	Continued Wet Storage	Melt and Dilute	Melt and Dilute[b]	Melt and Dilute[b]	Conventional Processing
TC E. Higher Actinide Targets	Continued Wet Storage	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site[c]	Continued Wet Storage	Repackage and Prepare to Ship to Another DOE Site[c]
F. Non-Aluminum-Clad Fuels	Continued Wet Storage	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site	Repackage and Prepare to Ship to Another DOE Site

a. Conventional processing would be the preferred technology for the failed or sectioned Oak Ridge Reactor fuel, High Flux Isotope Reactor fuel, Tower Shielding Reactor fuel, Heavy Water Components Test Reactor fuel, and a Mark-14 target.
b. Conventional processing is the preferred technology for the Sterling Forest Oxide fuel.
TC c. Conventional processing is the applicable technology for the Mark-18 target assemblies (approximately 1 kilogram heavy metal), under these two alternatives.

(DOE notes there is a high degree of technical uncertainty regarding the acceptability of this material in a repository; however, Direct Co-Disposal was postulated to represent minimum impacts.)

DOE would continue to wet store the Mark-51 and other Higher Actinide Targets at the SRS. Additionally, DOE would continue to wet-store the non-aluminum-clad spent nuclear fuel at SRS until the material is shipped to the Idaho National Engineering and Environmental Laboratory. In the event the non-aluminum clad fuel have not been transferred offsite by the time a dry storage facility is in operation at the SRS (to support the Melt and Dilute Technology), DOE could repackage the fuel and transfer the material to dry storage. To maintain operational flexibility, DOE could transfer the Mark-51 and other targets to dry storage. DOE would maintain the Mark-18 targets in wet storage pending disposition decisions due to potential health and safety concerns associated with the actions that would be required to repackage the Mark-18 target assemblies.

While in wet storage, if fuel began to deteriorate, resulting in imminent environmental, safety, and health vulnerabilities, DOE would use the canyons, if they were operating, to stabilize the vulnerable materials.

The loose uranium oxide in cans would not be contained in a tightly bound matrix and, therefore, may not be acceptable for placement in a geologic repository. Therefore, the Melt and Dilute technology would be used to treat these fuels.

2.4.2 MAXIMUM IMPACT ALTERNATIVE

This alternative provides the upper bound on the range of impacts from potential configurations. It would provide conventional processing for all SNF except the higher actinide targets and the non-aluminum-clad fuels selected for offsite shipment.

DOE expects that the Experimental Breeder Reactor-II and Mark-42 targets from the uranium and thorium metal fuels group would be processed in F Canyon. All other processing operations would be conducted in H Canyon. Processing operations in H Canyon would continue until the aluminum-based SNF inventory was eliminated and the SNF receipt rate was low (i.e., about 150 Materials Test Reactor-like elements per year and 12 High Flux Isotope Reactor assemblies per year; approximately 2009). In parallel with processing operations, DOE could construct a Transfer, Storage, and Treatment Facility with treatment capability to receive and treat new SNF after processing operations cease. Once the Transfer, Storage, and Treatment Facility was completed, processing in the canyons would be phased out.

Analyses of the maximum impact alternative are conservative in that they assume that the entire SNF inventory would be processed in the canyons, which would produce the greatest impacts of all the treatment options. No credit is taken for discontinuing use of the canyons and processing some of the inventory in a new treatment facility.

Although this EIS proposes only to continue to store Mark-18 targets, DOE has included the impacts of processing the Mark-18 targets in the Maximum Impact Alternative. The analysis of impacts is taken from the Final Environmental Impact Statement for Interim Management of Nuclear Materials. The 12-foot long Mark-18 targets would require size reduction for transport or storage in a dry storage facility. The standard method to reduce the size of the Mark-18 targets would be to cut them up under water in an SRS storage basin. The condition of the Mark-18 targets presents a health and safety vulnerability for under water cutting because of the suspected brittle condition of the targets and the uncertainty concerning which portion of the target assemblies contains the americium and curium product and fission products. Because of these concerns a previous DOE assessment (see Section 1.6.2) concluded that the Department should consider processing the Mark-18 targets. Although that

alternative was not chosen, and the Mark-18 targets are still stored in the Receiving Basin for Offsite Fuel, the analysis was performed and is incorporated as part of the Maximum Impact Alternative in this EIS. Processing the Mark-18 targets would not extend the operating time for the SRS canyons.

Until the Mark-51 and other Higher Actinide Targets are transferred to another site for use, DOE would continue to wet-store the material at the SRS. Additionally, DOE would continue to wet-store the non-aluminum-clad spent nuclear fuel at SRS until the material is shipped to the Idaho National Engineering and Environmental Laboratory. In the event the Mark-51 and "other" targets and non-aluminum clad fuel have not been transferred offsite by the time a dry storage facility is in operation at the SRS, DOE could repackage the targets and the fuel and transfer the material to dry storage. DOE would transfer the targets and non-aluminum clad fuel to dry storage after the material had been relocated from the Receiving Basin for Offsite Fuel to the L-Reactor Disassembly Basin in support of activities to phase out operations in the Receiving Basin for Offsite Fuel by 2007.

2.4.3 PREFERRED ALTERNATIVE

Under the preferred alternative, DOE would implement several of the technologies identified in Section 2.2 to manage spent nuclear fuel at SRS. These technologies are Melt and Dilute, Conventional Processing, and Repackage and Prepare to Ship. Each of these technologies would treat specific groups of spent nuclear fuel, as described below. The technology and fuel group combinations form DOE's Preferred Alternative in this EIS. The configuration of this preferred alternative is identified in Table 2-9. Figure 2-15 provides a flowchart for the Preferred Alternative.

2.4.3.1 Melt And Dilute

DOE has identified the Melt and Dilute process as the preferred method of treating most (about 97 percent by volume or about 32,000 MTRE) of the aluminum-based SNF considered in this EIS. DOE will continue to pursue a research and development program leading to a demonstration of the technology in FY 2001 using full-size irradiated research reactor spent nuclear fuel assemblies. With a successful demonstration of the technology, DOE expects to have ready a treatment facility to perform production melt and dilute operations in FY 2008. DOE will ensure the continued availability of SRS conventional processing facilities until we have successfully demonstrated implementation of the Melt and Dilute treatment technology.

The fuel proposed for the preferred Melt and Dilute technology includes the Material Test Reactor-like fuel, most of the Loose Uranium Oxide in Cans fuel, and most of the HEU/LEU Oxide and Silicide fuel. Exceptions are the failed and sectioned Oxide and Silicide fuel, about 10 percent of the Loose Uranium Oxide in Cans fuel as described in Section 2.4.3.2, and the Higher Actinide Targets and Non-Aluminum-Clad fuel that would be repackaged and prepared to ship as discussed in Section 2.4.3.3. The Melt and Dilute Technology satisfies DOE's objective and preference, as stated in the Record of Decision for the Nonproliferation Policy and Spent Nuclear Fuel EIS (60 FR 25091), to select a non-chemical separations-based technology to prepare aluminum-based SNF for placement in a geologic repository. Additionally, this new technology provides significant waste reduction (of high-level, low-level, transuranic, etc.) in comparison to conventional chemical processing and is fully compatible with and supportive of the nonproliferation objectives of the United States.

The potential impacts (e.g., worker and public health, waste generation, socioeconomics, etc.) among the new non-separations based technologies were all very similar; however, the Melt and Dilute option was the most efficient in volume reduction and produced the fewest number of SNF canisters. In fact, Melt and Dilute would increase volume reduction by more than 3 to 1 over Direct Disposal/Direct Co-Disposal. The volume reduction is achieved because the melt and dilute process eliminates voids in the fuel elements and in the canisters and fuel

Table 2-9. The fuel group technology configurations that compose the preferred alternative.

Fuel group	1. Prepare for Direct Co-Disposal	2. Repackage and Prepare to Ship[a]	3. Melt and Dilute	4. Mechanical Dilution	5. Vitrification Technologies	6. Electro-metallurgical Treatment	7. Conventional Processing
A. Uranium and Thorium Metal Fuels	–	–	–	–	–	–	Preferred
B. Materials Test Reactor-Like Fuels	–	–	Preferred	–	–	–	–
C. HEU/LEU[b] Oxides and Silicides Requiring Resizing or Special Packaging	–	–	Preferred	–	–	–	Preferred[c]
D. Loose Uranium Oxide in Cans	–	–	Preferred	–	–	–	Preferred[d]
E. Higher Actinide Targets[e]	–		–	–	–	–	–
F. Non-Aluminum Clad Fuels	–	Preferred	–	–	–	–	–

a. This alternative describes shipment to a DOE site other than SRS, not to a geologic repository.
b. HEU = highly enriched uranium; LEU = low enriched uranium.
c. For failed or sectioned Oak Ridge Reactor fuel, High-Flux Isotope Reactor fuel, Tower Shielding Reactor fuel, Heavy Water Components Test Reactor Fuel, and a Mark-14 target (i.e., <1 percent of material in this fuel group).
d. For Sterling Forest Oxide fuel (i.e., about 10 percent of the material in this fuel group).
e. The preferred alternative is to maintain fuel Group E in continued wet storage until a decision is made on final disposition.

Figure 2-15. Preferred Alternative Management Flow-Path.

baskets used in the Direct Disposal/Direct Co-Disposal technology. DOE considered Melt and Dilute to be among the most "proven" of the new non-separations-based technologies because DOE has made extensive progress in the development of the melt and dilute process.

The Melt and Dilute technology offers DOE the flexibility to engineer the final waste form to provide a high degree of confidence the material would be acceptable for placement in a geologic repository. Major technical concerns such as fuel characterization, criticality control, and repository performance can be reduced or eliminated by tailoring the chemical and physical form of the final product to meet specific criteria. DOE expects the Melt and Dilute option would be relatively simple to implement and would be less expensive than other similar technology options, although the ongoing technology development initiative will determine the viability of this alternative. The major technical issue for implementing this technology would be the design of an off-gas system to capture volatilized fission products. Preliminary engineering studies indicate that the system could be designed using proven approaches for managing off-gases.

TC

To implement the preferred alternative (Melt and Dilute technology), DOE would construct a melt and dilute facility in the existing 105-L building at SRS and build a dry-storage facility in L Area, near the 105-L building. DOE is proposing to use an existing facility to house the Melt and Dilute process because the existing structure can accommodate the process equipment and systems; the applicable portions of the structure will meet DOE requirements for resistance to natural hazards (e.g., earthquakes); the integral disassembly basin has sufficient capacity for all expected SNF receipts and the current Site inventory; using 105-L avoids the creation of a new radiologically controlled facility that would eventually require decontamination and decommissioning; and DOE has estimated the cost savings versus a new facility to be about $70 million.

EC

Using the Melt and Dilute technology, DOE would melt aluminum-based SNF and blend down any highly enriched uranium to low enriched uranium using depleted uranium that is currently stored at SRS. The material would be cast as ingots that would be loaded into stainless-steel canisters approximately 10 feet tall and 2 feet (or less) in diameter. The canisters would be placed in dry storage pending shipment to a geologic repository.

TC

During the development of the Melt and Dilute technology, DOE may determine that, for technical, regulatory, or cost reasons, the Melt and Dilute option is no longer viable. As a back-up to Melt and Dilute, DOE would continue to pursue the Direct Co-Disposal option of the New Packaging Technology and would implement this option if Melt and Dilute were no longer feasible or preferred. Direct Co-Disposal has the potential to be the least complicated of the new technologies and DOE believes this option could be implemented in the same timeframe as could the Melt and Dilute option. However, DOE believed there is greater risk in attempting to demonstrate that aluminum-based SNF, packaged according to the Direct Co-Disposal option, would be acceptable in a geologic repository. A comparison of the preferred (Melt and Dilute) and back-up (Direct Co-Disposal) technologies DOE proposes to use to manage most of the aluminum-based SNF at SRS is presented in Table 2-3.

TC

EC

If DOE identifies any imminent health and safety concerns involving any aluminum-based SNF, DOE could use F and H Canyons to stabilize the material of concern prior to the melt and dilute facility becoming operational.

TC

2.4.3.2 Conventional Processing

DOE proposes to use conventional processing to stabilize some materials before a new treatment facility is in place. The rationale for this processing is to avoid the possibility of urgent future actions, including expensive recovery actions that would entail unnecessary radiation exposure to workers, and in one case, to manage a unique waste form (i.e., core filter block).

TC

The total amount proposed for conventional processing is a relatively small volume of aluminum-based SNF at the SRS (about 3 % by volume and 40 % by mass). This material includes the Experimental Breeder Reactor-II fuel, the Sodium Reactor Experiment fuel, the Mark-42 targets and the core filter block from the Uranium and Thorium Metal fuel group; the failed or sectioned Tower Shielding Reactor, High Flux Isotope Reactor, Oak Ridge Reactor, and Heavy Water Components Test Reactor fuels and a Mark-14 target from the HEU/LEU Oxides and Silicides fuel group; and the Sterling Forest Oxide (and any other powdered/oxide fuel that may be received at SRS while H Canyon is still in operation) from the Loose Uranium Oxide in Cans fuel group. Although it is possible that a new treatment technology, such as melt and dilute, could be applied to most of these materials, DOE considers timely alleviation of the potential health and safety vulnerabilities to be the most prudent course of action because it would stabilize materials whose forms or types pose a heightened vulnerability to releasing fission products in the basin. Nonetheless, if these materials have not been stabilized before a new treatment technology becomes available, that new technology (melt and dilute) may be used rather than conventional processing.

The Experimental Breeder Reactor-II fuel and Sodium Reactor Experiment fuel are uranium metal that has been declad and stored in canisters in the Receiving Basin for Offsite Fuel. The declad fuels present a potential health and safety vulnerability. Should their existing storage containers leak, the metal fuel would corrode and release fission products to the water of the storage basin. Once the metal of the fuel is wetted, simply repackaging the fuel in a water-tight container would not arrest the corrosion and, in fact, could exacerbate storage concerns since potentially explosive hydrogen gas would continue to be generated inside the storage canister as the fuel continued to corrode. An instance of water intrusion and subsequent fuel corrosion has already occurred with one Experimental Breeder Reactor-II canister stored in the Receiving Basin for Offsite Fuel. Additionally, several problems have occurred with other uranium metal fuel in similar storage conditions at SRS (e.g., the Taiwan Research Reactor fuel with failed or missing cladding that was overpacked in canisters and stored in SRS wet basins). DOE addressed these situations by processing the failed or declad fuel in F Canyon to eliminate the health and safety vulnerability.

The failed or sectioned Tower Shielding Reactor, High Flux Isotope Reactor, Oak Ridge Reactor, and Heavy Water Components Test Reactor fuel, and a sectioned Mark-14 target from the HEU/LEU Oxides and Silicides fuel group also present potential health and safety vulnerabilities. The integrity of these fuels was destroyed for research purposes. Then the material was canned and placed in wet storage at SRS. A breach of or leak in the cans would expose the interior surfaces of the sectioned fuel to water, contaminating the water in the storage basin with radioactivity, and accelerating the corrosion of the fuel.

A potential health and safety vulnerability also exists for the unirradiated Mark-42 targets from the Uranium and Thorium Metal fuel group and the Sterling Forest Oxide fuel from the Loose Uranium Oxide in Cans fuel group. Should a breach occur in the cladding on the Mark-42 targets or in the canisters of Sterling Forest Oxide fuel, the particulate nature of the nuclear material in the targets and the Sterling Forest Oxide fuel could lead to dispersion of radioactive material in the water of the Receiving Basin for Offsite Fuel. Therefore, DOE is proposing to take action now to avoid the possibility of urgent future actions, including expensive recovery actions that also would entail unnecessary radiation exposure to workers.

DOE proposes to process the Experimental Breeder Reactor-II fuel and the Mark-42 targets in F Canyon. That fuel contains plutonium, approximately 114 kg of which would be recovered as part of the normal F Canyon chemical separations process and then transferred to FB-Line for conversion to metal. The plutonium metal would be considered surplus to the nation's nuclear

weapons program and would be placed in storage at the SRS pending disposition pursuant to the January 2000 Record of Decision (ROD) for the Surplus Plutonium Disposition Environmental Impact Statement (DOE 1999). The surplus plutonium would be immobilized using the can-in-canister process or fabricated into mixed-oxide (MOX) commercial power reactor fuel at the SRS. DOE has scheduled processing of the Experimental Breeder Reactor-II fuel and the Mark-42 targets in FY00.

DOE proposes to process the Sodium Reactor Experiment fuel, the failed or sectioned fuel from the HEU/LEU Oxides and Silicides fuel group, and the Sterling Forest Oxide fuel in H-Canyon where the highly enriched uranium would be blended down to low enriched uranium and stored pending potential sale as feed-stock for commercial nuclear fuel. DOE would begin processing operations in H Canyon in 2000 and could complete them in about 18 months.

TC

DOE also proposes to process the core filter block from the Uranium and Thorium Metals fuel group. The core filter block is made of depleted uranium but it contains corrosion-resistant metal (e.g., stainless-steel) that would be incompatible with the Melt and Dilute Technology for aluminum-based SNF. The core filter block could be processed in either F Canyon or H Canyon. In either case, the material would become feedstock to blend down highly enriched uranium from either conventional processing or melt and dilute operations.

The processing operations described above in both F and H Canyons would occur when the canyons were being operated to stabilize other nuclear material. It is the preference of the Department of Energy not to utilize conventional reprocessing for reasons other than safety and health. However, the core filter block is not compatible with the melt and dilute process for aluminum-based SNF. The benefit to develop a new process to accommodate this form would be disproportionately small when compared to the cost (DOE 1998a). Consequently, the Department proposes an exception in this case.

2.4.3.3 Repackaging

DOE would continue to wet-store the non-aluminum-clad spent nuclear fuel at SRS until the material is shipped to the Idaho National Engineering and Environmental Laboratory. In the event that the non-aluminum-clad fuel has not been transferred offsite by the time a dry storage facility is in operation at the SRS (to support the Melt and Dilute Technology), DOE could repackage the fuel and transfer the material to dry storage.

TC

2.4.3.4 Continued Wet Storage

DOE is not proposing any actions that would lead to the programmatic use of the higher actinide targets. Therefore, under the preferred alternative the Mark-18, Mark-51 and other higher actinide targets would be maintained in wet-storage until decisions are made on their final disposition.

2.4.4 DIRECT DISPOSAL ALTERNATIVE

This alternative combines the New Packaging and the New Processing Technologies with the Conventional Processing Technology. Materials Test Reactor-like fuels and HEU/LEU Oxides and Silicides (except the failed and sectioned fuels) would be treated using the Direct Disposal/Direct Co-Disposal technology and placed in the Transfer and Storage Facility with a minimum of treatment (e.g., cold-vacuum drying and canning).

DOE would manage the Higher Actinide Targets and the non-aluminum based SNF as described in the Maximum Impact Alternative.

TC

The uranium fuel and thorium metal fuel, Sterling Forest Oxide fuel from the Loose Uranium Oxide in Cans fuel group, and failed and sectioned fuel from the HEU/LEU Oxides and Silicides fuel group would be treated using chemical separations processes under the Conventional Processing Alternative to alleviate the potential

health and safety vulnerabilities discussed in Section 2.4.3.2 and because this material probably would not be suitable for placement in a geologic repository if treated with the Direct Disposal/Co-Disposal option. Most of the material in the Loose Uranium Oxide in Cans fuel group would be treated using Melt and Dilute since that material could be received after a melt and dilute facility was available.

2.4.5 NO-ACTION ALTERNATIVE: CONTINUED WET STORAGE

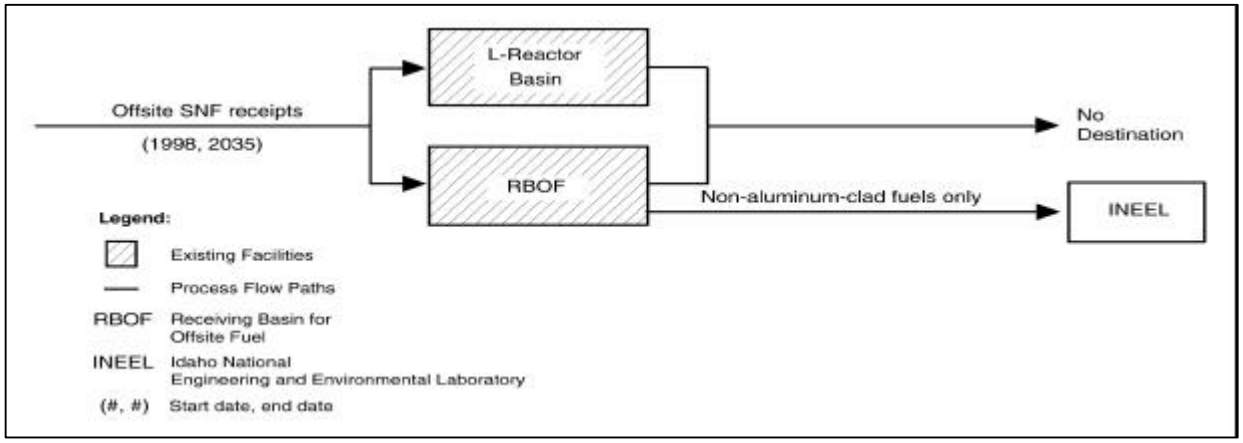
Under the No-Action Alternative, DOE would consolidate existing inventories of SNF at SRS in the L-Reactor Disassembly Basin and the Receiving Basin for Offsite Fuel, and would store incoming SNF shipments in those basins. Maintenance, monitoring, and normal basin operations (as described in Section 2.3.1) would continue. DOE would be able to meet its commitments to receive SNF from domestic, foreign, and university research reactors and from the Idaho National Engineering and Environmental Laboratory. However, DOE would not meet the commitment made in the Record of Decision (61 FR 25092) for the Final EIS on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (DOE 1996c) to manage its SNF in a road-ready condition for ultimate shipment to the geologic repository. DOE could ship non-aluminum-clad fuels to the Idaho National Engineering and Environmental Laboratory in accordance with the Record of Decision (60 FR 28680) for the Programmatic SNF EIS (DOE resulting in increased environmental, safety, and health vulnerabilities. DOE would use the F or 1995b). Over the potentially 40 years of continued wet storage, some fuel could deteriorate, H Canyon facilities if they were operating for other reasons to stabilize any SNF that presented an environmental, safety, or health vulnerability. Figure 2-16 shows the No-Action Alternative.

EC

DOE analyzed the impacts of transporting aluminum-based spent nuclear fuel to the Savannah River Site in the Nonproliferation Policy and Spent Nuclear Fuel EIS (DOE 1996c) and the programmatic SNF EIS (DOE 1995b). These documents concluded that the potential human health impacts from transportation of this fuel to SRS were low.

EC

Figure 2-16. No-Action Alternative – Continued Wet Storage.



TC

The No-Action Alternative would be applicable to all fuel groups; however, non-aluminum-clad fuels would remain in wet storage at SRS only until DOE shipped them to the Idaho National Engineering and Environmental Laboratory in accordance with the Programmatic SNF EIS Record of Decision. TC

2.4.6 ALTERNATIVES NOT ANALYZED IN DETAIL

DOE considered dry storing aluminum-based SNF (with no treatment or packaging) as a possible alternative for evaluation in this EIS. The first step for dry storing aluminum-based SNF would be accomplished by constructing a dry transfer facility. Fuel would be removed from wet storage in transfer casks, transported to the dry transfer facility, and removed from the transfer cask. Then the fuel would be placed in dry storage without any characterization, repackaging, or treatment that would be done under the New Packaging Technology alternative or New Processing Technology alternative. DOE decided not to evaluate this alternative because it would not meet the purpose and need for agency action (i.e., it would not prepare SNF for placement in a geologic repository). In order to prepare fuel for disposition, DOE would still have to implement the New Packaging Technology, New Processing Technology, or Conventional Processing alternatives, and dry storage is already analyzed as a component of these alternatives as applicable. EC TC

DOE considered a variation to the Chemical Processing Technology option where the dissolved Experimental Breeder Reactor-II fuel would be transferred to the high-level waste tanks at the SRS for subsequent vitrification in the Defense Waste Processing Facility. DOE evaluated this action under the *Interim Management of Nuclear Materials Final Environmental Impact Statement* (DOE 1995c) for material that is very similar to the Experimental Breeder Reactor-II fuel (i.e., Mark-31 targets and Taiwan Research Reactor SNF). In that EIS, DOE concluded that the process of transferring more than trace quantities of fissile material to the high-level waste tanks with subsequent vitrification was technically very complex and that it would take at least 6 years to develop the process. DOE noted that the Department would have to develop a process that would render fissile materials incapable of producing a nuclear criticality, regardless of the location or amount accumulated in various equipment or tanks. DOE postulated that this could be accomplished by the addition of a chemical or other material to serve as a nuclear "poison," which would minimize the potential for a criticality. However, the nuclear poison would have to be designed to accompany the fissile material throughout the process or different poisons would have to be used at different process steps (evaporation, concentration, precipitation, and ultimately vitrification). For these reasons, DOE does not consider this technology/fuel option reasonable for analysis in this EIS. Instead, DOE has analyzed the Dissolve and Vitrify option in the EIS, which would accomplish the same purpose as transferring the dissolved Experimental Breeder Reactor-II fuels to the high-level waste tanks for vitrification in the Defense Waste Processing Facility.

2.5 Comparison of Environmental Impacts Among Alternatives

Chapter 4 presents the predicted operational impacts, potential accident impacts, and construction impacts for each technology option and alternative. This organization enables the evaluation of recurring impacts (i.e., impacts from normal operations) independent of the infrequent impacts of accidents and the one-time impacts of construction.

As discussed in Section 1.3, DOE believes the amount of foreign research reactor SNF to be received in the U.S. could decrease from about 18 metric tons heavy metal (MTHM) to about 14 MTHM (or less). Therefore, the actual amount of aluminum-based material could be less than the 48 MTHM evaluated in this EIS. The only effect would be a small reduction of environmental impacts described in this EIS. DOE does not believe a reduction of this magnitude would materially affect the impacts associated EC

with normal operations involving Material Test Reactor-like fuels (Fuel Group B) and the reduction would occur across all alternatives. However, where it is applicable, DOE has included information in the impact tables for normal operations that provide an example of how the reduced Fuel Group B impact data could be calculated. EC

The potential reduction in foreign research reactor SNF receipts would have no effect on the accident impact data that are presented in the EIS because none of the postulated accidents could affect all the fuel at once. Processing related accidents would affect only the "batch" of fuel that was involved in the process operation and accidents that could affect stored fuel, such as an earthquake, would be unlikely to involve all the fuel in the storage facility.

Impacts from normal operations under all of the alternatives would have little if any effect on ecological resources, water resources, or cultural resources. The impacts from incident-free onsite transportation of SNF would be minimal under all alternatives.

Processing the Mark-18 targets (about 1 kilogram of heavy metal) was previously analyzed in the *Final Environmental Impact Statement on Interim Management of Nuclear Materials* and, therefore, was not analyzed in this EIS. The impacts of processing this small amount of material are minor and would not significantly affect the impacts analyzed for the Maximum Impact Alternative in this EIS. For example, total radiological dose from the Preferred Alternative to the maximally exposed individual for the entire period of analysis would be 0.67 millirem. Processing the Mark-18 targets would result in a dose of 0.0035 millirem. TC

Table 2-10 lists impacts for the five selected alternatives. The EIS identifies the following operational impacts with the potential to discriminate among the alternatives:

- W*orker and public health impacts* – Estimated impacts are reported as latent cancer fatalities for the involved worker population, noninvolved worker, the maximally exposed member of the public, and offsite population. These impacts are summed over the period of analysis based on annual emissions and radiation doses.

  Involved worker doses assume that no worker would receive more than the SRS administrative annual limit of 700 millirem. Based on this, the estimated latent cancer fatalities for the involved worker population for the entire period of analysis would range from 0.28 for the Minimum Impact Alternative to 0.84 for the Maximum Impact Alternative.

  The values in Table 2-10 for health effects to the noninvolved worker, maximally exposed individual, and the offsite population for the No-Action Alternative represent current reactor-area emissions (including two SNF wet basins) for the entire period of analysis. The values for the other alternatives would be incremental above these baseline values. Summing these baseline and incremental values would be conservative, however, because there would not be two SNF wet basins operating over the entire 38-year period of analysis.

  The noninvolved worker highest estimated probability of a latent cancer fatality over the entire period of analysis would range from 2.0×10^{-9} for the Minimum Impact Alternative to 6.3×10^{-7} for the Maximum Impact Alternative.

  The estimated latent cancer fatality probability to the maximally exposed individual over the entire period of analysis would range from 3.0×10^{-10} (Minimum Impact Alternative) to 3.4×10^{-7} (Maximum Impact Alternative). The estimated latent cancer fatalities in the offsite population affected by SRS over the entire period of analysis would be much less than 1 for any alternative. These estimated offsite latent cancer

Table 2-10. Impact summary by alternative.

Parameter	No Action Alternative (baseline)	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative
Health Effects for the Entire Period of Analysis (1998-2035)					
TC Latent cancer fatality probability for the noninvolved worker	1.7×10^{-6}[a]	2.0×10^{-9}	9.6×10^{-9}	6.1×10^{-7}	6.3×10^{-7}
Latent cancer fatality probability for the maximally exposed member of the public	3.1×10^{-7}[a]	3.0×10^{-10}	3.6×10^{-9}	9.5×10^{-8}	3.4×10^{-7}
Latent cancer fatalities for the worker population	0.30	0.28	0.34	0.33	0.84
Latent cancer fatalities for the general public	1.1×10^{-2}[a]	1.1×10^{-5}	3.8×10^{-5}	3.4×10^{-3}	4.4×10^{-3}
Waste Generation Required for the Entire Period of Analysis (1998-2035)					
TC Liquid (cubic meters)	2,300	660	1,200	1,050	10,500
High-level waste generated (equivalent DWPF[b] canisters)	38	11	20	17	160
Transuranic waste generated (cubic meters)	0	15	360	563	3,700
Hazardous and mixed low-level waste generated (cubic meters)	76	25	46	103	267
Low-level waste generated (cubic meters)	57,000	20,000	31,000	35,260	140,000
Utilities and Energy Required for the Entire Period of Analysis (1998-2035)					
Water consumption(millions of liters)	1,100	660	1,400	1,186	8,000
Electricity consumption (megawatt-hours)	46,000	27,000	81,000	116,000	600,000
Steam consumption (millions of kilograms)	340	190	520	650	3,600
Diesel fuel consumption (thousands of liters)	230	180	2,300	2,760	22,000
Road-ready Repository canisters (1998-2035)	0	~1,400	~1,300	~400	0[c]

a. Reflects current reactor-area emissions (including two SNF wet basins) for the entire period of analysis.
b. DWPF = Defense Waste Processing Facility.

c. The technology used in the Maximum Impact Alternative (i.e., Conventional Processing) would produce only high-level waste.

Table 2-11. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for each fuel group and technology (percent of regulatory standard).

	Technology						
Fuel group	1. Prepare for Direct Co-Disposal	2. Repackage and Prepare to Ship[a]	3. Melt and Dilute	4. Mechanical Dilution	5. Vitrification Technologies	6. Electro-metallurgical Treatment	7. Conventional Processing
A. Uranium and Thorium Metal Fuels	0.02 (ozone [as VOC])	NA	0.03 (ozone [as VOC])	No	1.1 (nitrogen oxides)	0.03 (ozone [as VOC])	1.1 (nitrogen oxides)
B. Materials Test Reactor-Like Fuels	0.03 (ozone [as VOC])	NA	0.05 (ozone [as VOC])	0.03 (ozone [as VOC])	1.7 (nitrogen oxides)	0.05 (ozone [as VOC])	1.7 (nitrogen oxides)
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0.01 (ozone [as VOC])	NA	0.02 (ozone [as VOC])	0.01 (ozone [as VOC])	0.55 (nitrogen oxides)	0.02 (ozone [as VOC])	0.55 (nitrogen oxides)
D. Loose Uranium Oxide in Cans	NA	NA	<0.004 (ozone [as VOC])	NA	0.06 (nitrogen oxides)	<0.002 (ozone [as VOC])	0.06 (nitrogen oxides)
E. Higher Actinide Targets	NA	<0.004 (ozone [as VOC])	NA	NA	NA	NA	NA
F. Non-Aluminum-Clad Fuels	NA	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.
VOC = volatile organic compound.

Table 2-12. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for each alternative (percent of regulatory standard).

No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative
0.03 (nitrogen oxides)	0.07 (ozone [as VOC])	1.2 (nitrogen oxides)	1.1 (nitrogen oxides)	3.6 (nitrogen oxides)

VOC = volatile organic compound.

fatalities would range from 1.1×10^{-5} to 4.4×10^{-3}.

· *Nonradiological Air Quality* – Table 2-11 presents the estimated maximum incremental concentrations of the nonradiological air pollutants that would contribute the most to the deterioration of air quality at the SRS boundary. Concentrations are presented for each technology fuel group concentration. The incremental concentrations would not affect human health. Table 2-12 presents the estimated maximum incremental concentration of the nonradiological air pollutant that would contribute the most to the deterioration of air quality at the SRS boundary for each alternative. As noted from Table 2-12, the concentration of the nonradiological constituent contributing the highest fraction of the offsite air quality standard would range from 0.03 percent of the standard for the No-Action Alternative to 3.6 percent of the standard for the Maximum Impact Alternative. Under all alternatives, nonradiological air concentrations of the SRS boundary would be well below applicable standards.

- *Waste generation* – Wastes volumes were estimated over the period of analysis. The Maximum Impact Alternative would generate the greatest volume of high-level waste, while the Minimum Impact Alternative would generate the least volume of high-level waste. For wastes generated under all alternatives, DOE would use the surplus capacity in existing SRS waste management facilities to treat, store, dispose, or recycle the waste in accordance with applicable regulations.

- U*tilities and energy consumption* – The quantities of water, electricity, steam, and diesel fuel that would be required over the entire period of analysis were estimated.

  The Maximum Impact Alternative would require the most water, electricity, steam, and diesel fuel, while the Minimum Impact Alternative would require the least. For all alternatives, water and steam would be obtained from existing onsite sources and electricity and diesel fuel would be purchased from commercial sources. These commodities are readily available and the amounts required would not have an appreciable impact on available supplies on capacities.

  Accidents – DOE evaluated the impacts of potential facility accidents related to each of the alternatives. For each potential accident, the impacts were evaluated as radiation dose to the noninvolved worker, radiation dose to the offsite maximally exposed individual, collective radiation dose to the offsite population, and latent cancer fatalities to the offsite population. Table 2-13 presents the results of this analysis. Table 2-13 also indicates the estimated frequency of occurrence for each accident.

  The highest consequence accident postulated under the continued wet storage, direct co-disposal, and repackage and prepare to ship technologies is a seismic/high wind-induced criticality, which is estimated to TC

Table 2-13. Estimated maximum consequence accident for each technology.

Option	Accident Frequency	Consequences: Noninvolved Worker (rem)	Consequences: MEI (rem)	Consequences: Offsite Population (person-rem)	Consequences: Latent Cancer Fatalities
Continued Wet Storage (No Action)[a]					
RBOF (high wind-induced criticality)	Once in 26,000 years	13	0.22	12,000	6.2
L-Reactor basin (basin-water draindown)	Once in 500 years	0.014	0.016	(b)	(b)
Direct Co-Disposal					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Repackage and Prepare to Ship					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Conventional Processing					
Processing phase in F/H Canyons (coil and tube failure)	Once in 14,000 years	13	1.3	78,000	39
Melt and Dilute					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Melt and dilute phase (earthquake induced spill with loss of ventilation)	Once in 200,000 years	30	0.5	21,000	10
Mechanical Dilution					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Mechanical dilution phase (criticality with loss of ventilation)	Once in 33,000 years	0.71	0.074	3,000	1.5
Vitrification Technologies					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Vitrification phase (earthquake-induced release with loss of ventilation)	Once in 200,000 years	0.10	0.0017	71	0.035
Electrometallurgical Treatment					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Electrometallurgical phase (metal melter earthquake induced spill with loss of ventilation)	Once in 200,000 years	30	0.5	21,000	10

MEI = Maximally Exposed Individual.
RBOF = Receiving Basin for Offsite Fuels.
a. All alternatives would use RBOF and the L-Reactor Disassembly Basin; therefore, accidents in these facilities are possible for each technology.
b. Not available.

result in 6.2 latent cancer fatalities in the offsite population. The highest consequence accident under conventional processing technology is a coil and tube failure with an estimated offsite population impact of 39 latent cancer fatalities. The frequencies of these accidents are once in 2,000 to once in 26,000 years.

TC

For the other new SNF technologies evaluated, the maximum consequence accident (earthquake induced spill with loss of ventilation) is associated with the melt and dilute process. This accident is estimated to occur once in 200,000 years and to result in 10 latent cancer fatalities in the offsite population.

Construction activities could affect four parameters: surface-water quality, air quality, ecological resources, and socioeconomics. However, because current SRS construction workers would build the facilities in an existing industrialized area of the Site, DOE expects little impact from construction activities.

2.6 Other Decisionmaking Factors

2.6.1 TECHNOLOGY AVAILABILITY AND TECHNICAL FEASIBILITY

The New Packaging and New Processing Technology Alternatives would rely on technologies that have not been applied to the management of aluminum-based SNF for ultimate disposition. Therefore, DOE conducted a feasibility study of the non-processing technologies and documented the study in a report prepared by a Research Reactor Task Team in its Office of Spent Fuel Management (DOE 1996b).

The Research Reactor Task Team examined a wide range of technical issues involved in achieving safe and cost-effective disposal of aluminum-based SNF under DOE jurisdiction. The Team identified and evaluated issues on technical grounds to arrive at a recommendedcourse of action that could lead to the implementation of a non-processing SNF management technology by 2000. The team considered three specific areas of investigation to be key: (1) repository and waste form considerations; (2) SNF receipt, handling, and storage provisions; and (3) treatment technologies (the same technologies this EIS considers). The team assigned the highest confidence of success and greatest technical suitability to technologies that would have relatively simple approaches (i.e., Direct Disposal, Direct Co-Disposal, Melt and Dilute, and Press and Dilute). The Conventional Processing option would have the least technical uncertainty because it would rely largely on a technology that is proven for aluminum-based SNF. The No-Action Alternative would involve the greatest technical uncertainty in the area of potential fuel degradation, as a result of continued long-term wet storage in SRS basins. The non-processing technologies with the greatest technical uncertainties would be the more complicated technologies such as vitrification.

L3-3

In response to a DOE request, the National Academy of Sciences evaluated and provided recommendations for DOE's aluminum-based SNF disposition technical program (NAS 1998). The NAS report was prepared by a Principal Investigator assisted by a panel of expert consultants in fields of nuclear criticality control, proliferation policy, costs and schedules, corrosion and metallurgy, processing and remote handling, and regulatory waste acceptance.

The panel reviewed the DOE program for developing a strategy for treatment of aluminum-based SNF in preparation for interim storage and final disposal, with emphasis on the following objectives:

- Evaluation of the set of technologies proposed by DOE for aluminum-based SNF treatment, with suggestions of other applicable technologies

- Examination of waste package performance criteria developed by DOE to meet the anticipated waste acceptance criteria for storage, transportation, and repository disposal

- Assessment of projected costs and schedule for implementation of the aluminum-based SNF technologies

The NAS report generally endorsed the projected DOE spent fuel disposition scenarios under development. The NAS recommendations for systems approach and phased strategy were incorporated by DOE into the EIS as follows:

Two systems analyses were completed for the primary new technologies being considered by DOE (Melt and Dilute and Direct Disposal/Direct Co-Disposal). A variety of attributes were evaluated, including cost, criticality concerns, public safety, worker safety, environmental concerns, nonproliferation, versatility, maintainability, and repository volume. One analysis was performed by Westinghouse Savannah River Company (WSRC 1998b), and a second independent multi-attribute decision analysis was completed by Sandia National Laboratory (SNL 1998). In both studies, Melt and Dilute had the least uncertainty.

DOE has recognized the advantages of applying a phased strategy for implementation of the melt and dilute process and continues to integrate its development and installation with other site program priorities and schedules in mind. The NAS concern regarding technology selection being driven by post-2015 SNF receipts is mitigated by the plan to design a facility with minimal-sized processing capabilities, which will be able to treat the current inventory of spent nuclear fuels within a reasonable timeframe, yet not be operationally burdensome when fuel receipts are reduced to minimal amounts.

TC

The phased strategy was accommodated by provisions of backup treatments for appropriate fuel types should the projected preferred treatments not be successfully implemented within required time constraints. For example, the Direct Disposal/Direct Co-disposal technology is included as a backup technology for Melt and Dilute technology.

In summary conclusions, the NAS noted the complexity of the aluminum-based SNF disposal program including factors such as: the timely provision of initial storage capacity for the fuel at SRS; the selection, development, and implementation of one or more treatment options to qualify the fuel for possible repository disposal; and the interim storage required until the repository, yet-to-be designed, licensed, or constructed, can accept it. The Academy noted that an SNF disposition program requires a systems approach for optimization of the many interacting factors required for successful implementation. The NAS recommended that aluminum-based SNF treatment decisions be made using a phased strategy in which critical decisions are made as the information needed for sound choices becomes available, recognizing the trade-offs between information acquisition and costs of delayed decisions.

The NAS panel identified a number of specific findings with recommendations as described in their report (NAS 1998).

Specific observations of the panel included the following:

- DOE has identified a reasonably complete set of aluminum-based SNF treatment options, resulting in selection of the Direct Co-Disposal and Melt and Dilute technologies for further development.

- The selection of a preferred treatment alternative must take into account uncertainties in repository Waste Acceptance Criteria that could, for example, disqualify highly enriched uranium waste forms such as produced by the Direct Co-Disposal technology.

- Both the Direct Disposal/Direct Co-Disposal and Melt and Dilute technologies apparently can be implemented to produce acceptable waste forms. The high-temperature Melt and Dilute treatment is technically more demanding than the relatively straight-forward Direct Disposal/ Direct Co-Disposal treatment and presents potential problems in ra-

dioactive off-gas control, but the basic operations have been demonstrated in other programs. Suitability of other technology options, such as the Electrometallurgical Treatment, is less assured because of the additional development work needed. EC

- More careful consideration of the conventional processing option is needed, because it is a well-demonstrated technology, its costs and risks are known, the necessary facilities are in current operations, and the high-level waste form is likely acceptable in the repository.

- DOE has established a working relationship with DOE-Yucca Mountain and plans to continue this relationship to ensure timely identification of repository waste form criteria and waste characterization requirements.

- Other waste form criteria, including interim-storage criteria, appear reasonable and complete, except for transportation requirements. The panel recommended DOE review shipping requirements before finalization of canister/shipping cask design for the waste forms.

- Work under way by DOE-SR appears properly focused and appropriate to the above requirements. However, a single treatment option may not be suitable for all types of aluminum-clad SNF and the program should maintain flexibility in technology selection to accommodate this variability.

- Major cost factors are accounted for in the cost projections, but schedule projections appear ambitious, and schedule delays could affect the cost projections. Projected costs are, however, not a major discriminator of the various treatments and treatment selection can proceed based on current projections.

The DOE-SR and the Nuclear Regulatory Commission (NRC) have established an agreement for the NRC to provide technical assistance in connection with the identification of potential issues relating to the placement of aluminum-based foreign and domestic research reactor spent nuclear fuel in a geologic repository. In a recent review of DOE's research and development work, the NRC staff indicated that both the Melt and Dilute and Direct Co-Disposal technologies would be acceptable concepts for the disposal of aluminum-based research reactor SNF in a repository (Knapp 1998).

2.6.2 NONPROLIFERATION, SAFEGUARDS AND SECURITY

On May 13, 1996, the United States established a new 10-year policy to accept and manage foreign research reactor spent nuclear fuel containing uranium enriched in the United States (61 FR 25091). The goal of this policy is to reduce civilian commerce in weapons-usable highly enriched uranium, thereby reducing the risk of nuclear weapons proliferation, as called for in President William Clinton's September 27, 1993, Nonproliferation and Export Control Policy.

Two key disposition options under consideration for managing SNF in this EIS include conventional processing and new treatment and packaging technologies. The Record of Decision for managing foreign research reactor SNF specified that, while evaluating the processing option, "DOE will commission or conduct an independent study of the nonproliferation and other (e.g., cost and timing) implications of chemical separation of spent nuclear fuel from foreign research reactors." DOE's Office of Arms Control and Nonproliferation conducted the study. To receive a copy, contact DOE at 1-800-881-7292.

The study addresses the nonproliferation implications the Department considered in determining how to manage aluminum-based SNF at the Savannah River Site, including how to place these materials in forms suitable for ultimate disposition (DOE 1998a). Because the same technology options are being considered for the foreign research reactor and the other aluminum-based spent nuclear fuels, the report addresses the nonproliferation implications of managing all the Savannah River Site aluminum-based SNF.

The nonproliferation assessment evaluates the extent to which each technology option supports the United States nonproliferation goals, which are summarized below.

- To reduce the risk of nuclear proliferation and for other considerations, the United States neither encourages the civil use of plutonium nor engages in plutonium processing for either nuclear power or nuclear explosive purposes. In addition, the United States works actively with other nations to reduce global stocks of excess weapons-usable material; separated plutonium and highly enriched uranium. Under this policy, the United States honors its commitments to cooperate with civilian nuclear programs that involve the processing and recycling of plutonium in Western Europe and Japan. In all such cases, however, the United States seeks to ensure that the International Atomic Energy Agency (IAEA) has the resources needed to implement its vital safeguards responsibilities, and works to strengthen the IAEA's ability to detect clandestine nuclear activities. The United States seeks to eliminate where possible the accumulation of stockpiles of highly enriched uranium or plutonium, and to ensure that where these materials already exist they are subject to the highest standards of safety, security, and international accountability. The United States also actively opposes, as do other supplier nations, the introduction of processing and plutonium recycling activities in regions of proliferation concern.

- The United States also seeks to minimize the adverse environmental, safety, and health impacts of its management of nuclear materials and activities. This goal includes minimizing the generation of radioactive wastes and ensuring that waste materials are put into forms that can be disposed of safely.

To evaluate the extent to which the technology options support the United States' nonproliferation policy goals, the nonproliferation study evaluated the technology options using technical and policy factors, as explained below.

Technical factors include the degree to which a particular technology would:

- Help ensure that the weapons-usable nuclear material in the spent nuclear fuel could not be stolen or diverted during the process. This includes an assessment of the attractiveness to diversion of materials in process and the ease of providing institutional and inherent security features.

- Facilitate cost-effective international verification and transparency.

- Result in converting the spent nuclear fuel into a form from which retrieval of the material for weapons use would be difficult and unlikely, thus modestly reducing the total stockpile of material readily usable in nuclear weapons.

Policy factors include the degree to which a particular technology would:

- Be consistent with United States policy related to processing and nonproliferation.

- Avoid encouraging other countries to engage in the processing of spent nuclear fuel, or undermining United States efforts to limit the spread of processing technology and activities, particularly to regions of proliferation concern.

- Support United States efforts to convert United States and foreign research reactors to low enriched fuels, and avoid creating technical, economic, or political obstacles to implementing the Foreign Research Reactor Spent Nuclear Fuel Acceptance Program.

- Help demonstrate that any treatment of these spent nuclear fuels will definitely not represent the production by the United States of additional materials for use in nuclear weapons.

- Support negotiation of a nondiscriminatory global fissile material cutoff treaty.

There are several options for the effective management of the aluminum-based SNF at SRS.

With respect to nonproliferation, the report concluded the following:

- All of the options could reliably discourage any theft or diversion of the material, but some are superior to others.

- All of the options could provide for some form of international safeguarding by the International Atomic Energy Agency (IAEA). The options vary in terms of cost and ease of application.

- All of the options would result in forms from which recovery of the material for use in weapons would be highly unlikely, although the Direct Disposal/Direct Co-Disposal Option would not blend down the residual highly enriched uranium and low enriched uranium, and the conventional processing option would recover plutonium metal that would be managed as surplus.

- All of the options would be consistent with United States nonproliferation policy, and would allow for verification approaches that would be acceptable to the United States if implemented in other countries.

- The electrometallurgical treatment and the conventional processing, by appearing to endorse these technologies, could conceivably encourage processing in other countries.

- All of the options have the potential to support fully United States efforts to reduce the civil use of highly enriched uranium, including the Foreign Research Reactor Spent Nuclear Fuel Acceptance Program.

- None of these options would appear to be prejudicial to the ability of the United States to submit to international safeguards or monitoring under a nondiscriminatory fissile material cutoff treaty. However, the processing option involves the use of old facilities at the Savannah River Site not specifically designed to facilitate the application of international safeguards. An effective safeguarding regime would likely be difficult due to cost and safety retrofitting concerns (DOE 1998a).

- The Office of Arms Control and Nonproliferation fully supports the active pursuit of a new treatment technology for the aluminum-based spent nuclear fuel, and views the melt and dilute recommendation as a favorable technology in light of nonproliferation concerns.

2.6.3 LABOR AVAILABILITY AND CORE COMPETENCY

Each alternative and associated technologies would require different levels of personnel knowledge and training. In addition, providing the needed level of training would result in impacts, primarily in the area of personnel resources. In general, the New Packaging options probably would be the least labor-intensive. The Conventional Processing option or a combination of options that included conventional processing would be the most labor-intensive to implement on an annual basis.

Operations required for the Conventional Processing technology would occur in parallel with other canyon nuclear stabilization programs. As a result, no excess personnel would be available in the event the vulnerable SNF was not processed. Because the canyons already would be operating to process materials not considered in this EIS, there also would be no actual cost savings that could be transferred to another activity. L2-12

The Conventional Processing technology option and No-Action Alternative would require the least amount of training because the SRS workforce has a great deal of experience in these technologies and there are existing training and qualification programs to maintain core compe-

tency. The New Processing Technology options such as Vitrification Technologies or Electrometallurgical Treatment probably would require the greatest training effort because they would involve new and complex operations.

2.6.4 MINIMUM CUSTODIAL CARE

The New Packaging Technology and New Processing Technology options would create a form of material that required the least amount of custodial care before shipment off the Site. However, safeguards and security requirements would still be maintained. Conventional processing would require care of the vitrified waste similar in level-of-effort to the custodial care of the New Packaging and New Processing Technology option. In addition, it also would require care of the high-level waste until it was vitrified and any blended-down fissile material until they were delivered for disposition.

2.6.5 COST

To determine the potential cost of integrating various combinations of alternatives, DOE has estimated life-cycle costs for the alternatives and for the new technology options described in this EIS and for conventional processing. The cost report was prepared, in part, to satisfy the Department's commitment to study the implications of chemically separating SNF (see Section 2.6.2). The planning level costs have an uncertainty of +50 percent to -30 percent. These estimates, which are listed in Table 2-14, include both operating and capital (i.e., construction) costs (DOE 1998b).

DOE estimated the costs for the alternatives discussed in this EIS using the technology option cost information from the cost study. The cost estimates for the alternatives are presented in Table 2-15.

Comparison of the projected life cycle costs for the alternatives indicate the following:

- The life-cycle costs range from a low of $1.7 billion for No Action to a high of $2.0 billion for the Maximum Impact Alternative. However, the continued wet storage cost does not include actions necessary to prepare SNF for ultimate disposition.

- The Direct Disposal Alternative ($1.9 billion) and the Preferred Alternative ($2.0 billion) (both using a renovated reactor building) have approximately the same life-cycle cost, with installation in a renovated reactor facility presenting cost advantages of about $200 million compared to a new treatment facility. EC

- The cost of processing the SNF proposed in the Preferred Alternative would be incremental to the cost of operating the canyons for other reasons and very small when compared to the canyon overall operating cost. L4-15

Table 2-14. Life-cycle costs for aluminum-clad fuel technologies (1998 millions of dollars)[a].

Table 2-15. Life-cycle costs (1998 billions of dollars) for each alternative.[a]

Minimum Impact	Direct Disposal	Preferred Alternative	Maximum Impact	No Action
1.9[b]	1.9	2.0[c]	2.0	1.7

EC

a. Source: DOE (1998b).
b. Includes less than $30 million to install Melt and Dilute capability for Fuel Group D.
c. Includes about $6 million as direct and indirect cost of operating canyons for SNF processing during 1999-2001 while the material stabilization program is underway in response to Defense Nuclear Facility Safety Board Recommendation 94-1.

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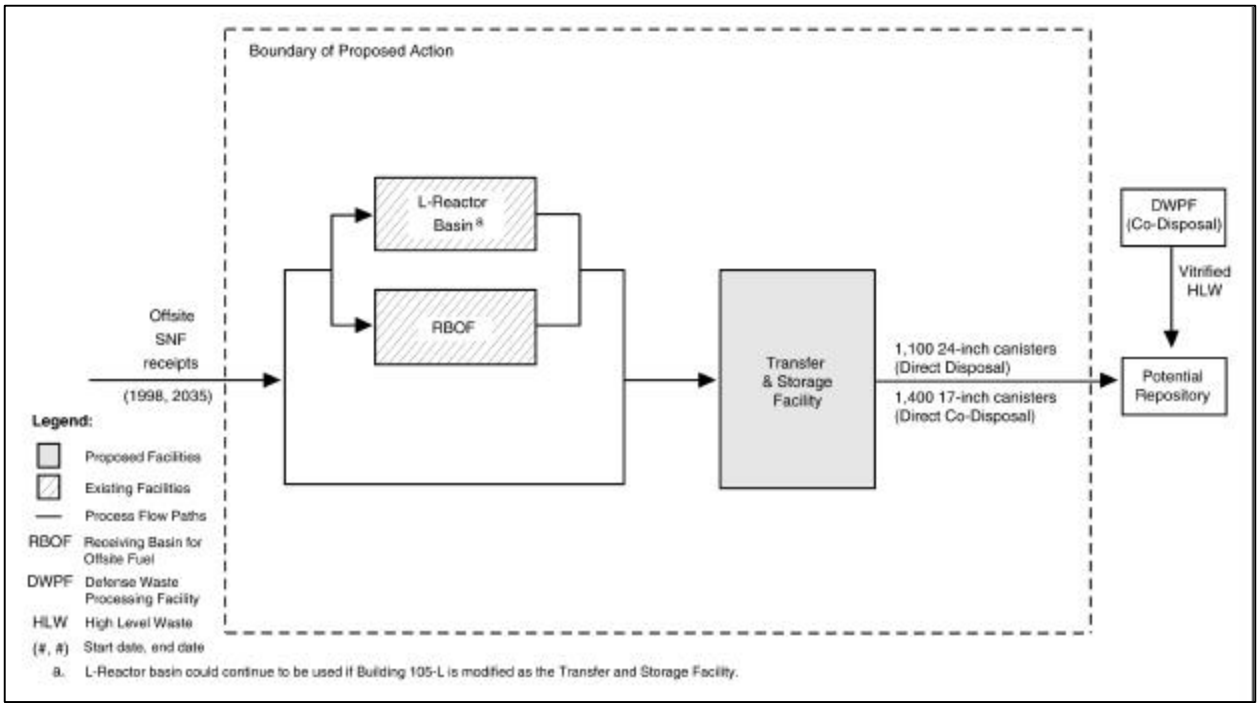

Figure 2-1. New Packaging Technology – Direct Disposal/Direct Co-Disposal.

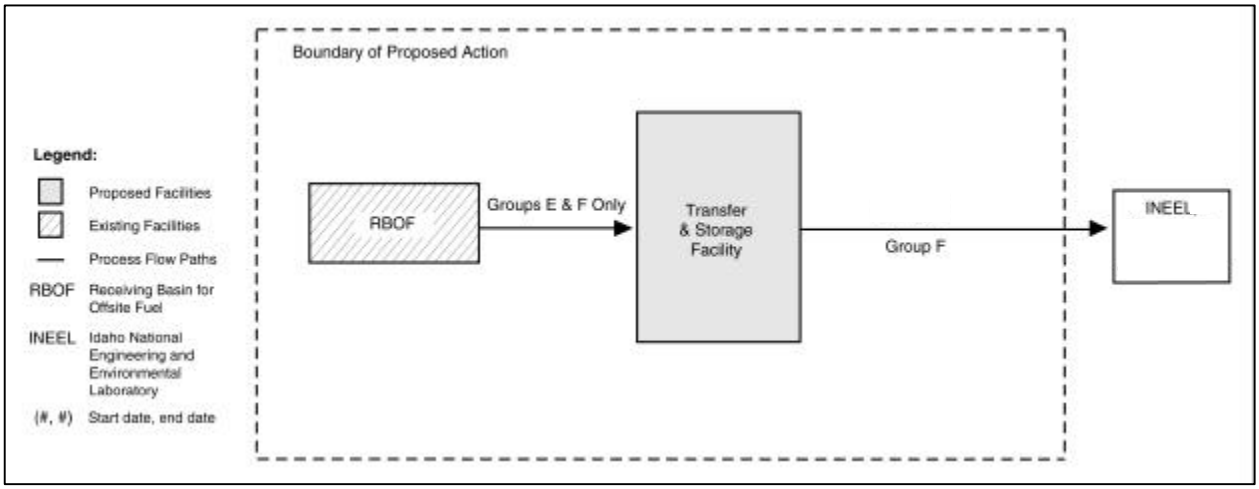

Figure 2-2. New Packaging Technology – Repackage and Prepare to Ship to Another DOE site.

Figures 2-3 and 2-4 show the New Processing Technology options. The following sections describe the new technology options; Appendix A describes them in more detail. Table 2-2 lists the applicability of the New Processing Technology to the fuel groups described in Chapter 1.

2.2.4.1 Melt and Dilute

Under the Melt and Dilute option, DOE would receive, unload, and crop the SNF in the Transfer, Storage, and Treatment Facility and either package the fuel in canisters for placement in dry storage pending treatment or send it directly to the treatment phase. The SNF would be melted and, if highly enriched, mixed with depleted ura-

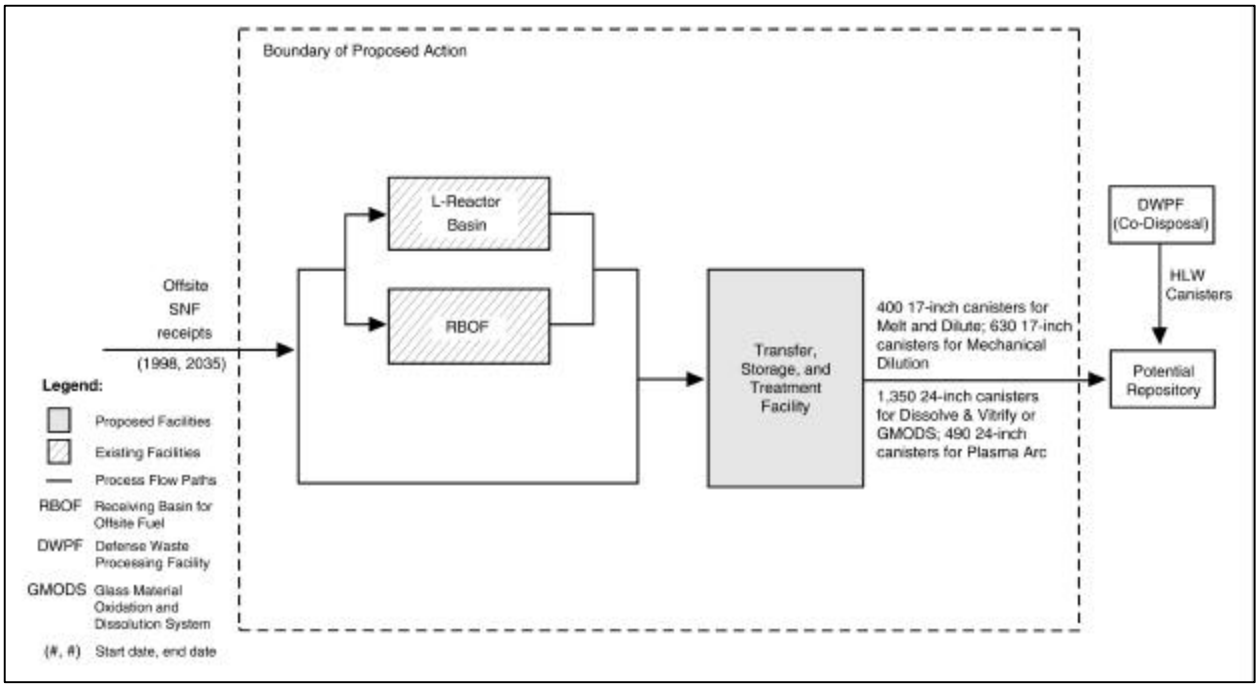

Figure 2-3. New Processing Technology - Melt and Dilute, Mechanical Dilution, Vitrification Technologies.

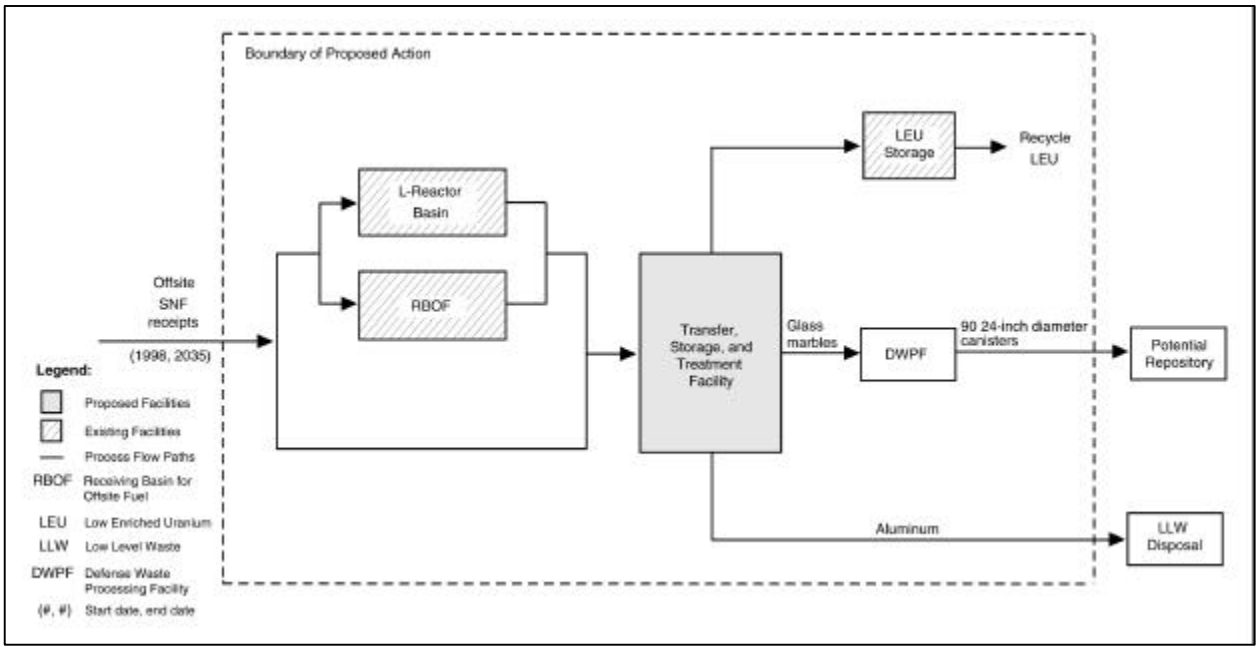

Figure 2-4. New Processing Technology – Electrometallurgical Treatment.

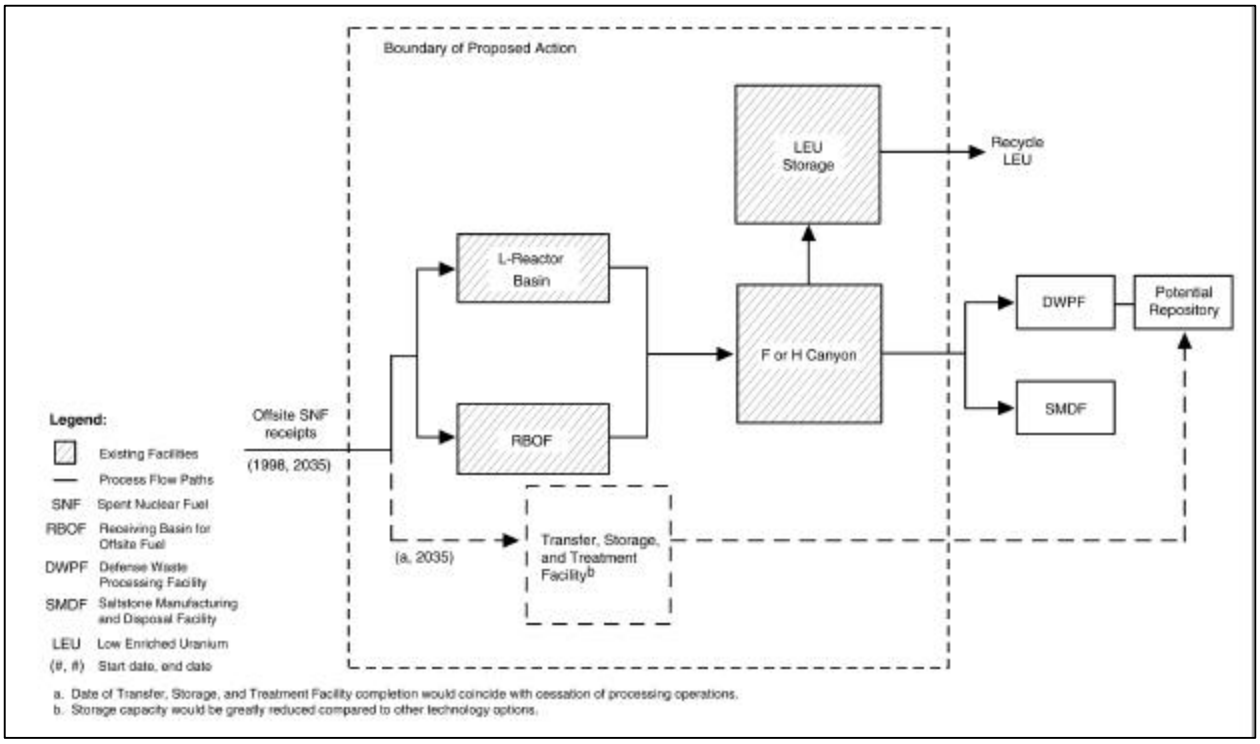

Figure 2-5. Conventional Processing.

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Table 2-4. Facilities needed for SNF technologies.

Technology	Receiving Basin for Offsite Fuel	L-Reactor Facility	F or H Canyon	Transfer and Storage Facility	Melt and Dilute Treatment Facility	Mechanical Dilution Treatment Facility	Vitrification Facility	Electrometallurgical Treatment Facility	Renovated Reactor Facility
1. Prepare for Direct Disposal/Direct Co-Disposal	✓	✓		✓					✓
2. Repackage and Prepare to Ship[a]	✓	✓[b]		✓					✓
3. Melt and Dilute	✓	✓		✓	✓				✓
4. Mechanical Dilution	✓	✓		✓		✓			✓
5. Vitrification Technologies	✓	✓		✓			✓		✓
6. Electrometallurgical Treatment	✓	✓		✓				✓	✓
7. Conventional Processing	✓	✓	✓	✓	✓[c]				
8. Continued Wet Storage	✓	✓							

a. To another DOE site.
b. Needed only if a Transfer, Storage, and Treatment Facility were implemented in a reactor facility.
c. Once conventional processing is terminated, the remaining SNF would require treatment using one of the new technologies. A Melt and Dilute Treatment Facility is included as part of Conventional Processing as a reference follow-on treatment

Table 2-14. Life-cycle costs for aluminum-clad fuel technologies (1998 millions of dollars).[a]

	Technology									
	1	1A	2	3	3A	4	5	6	7	8
Cost factors	Prepare for Direct Co-Disposal	Prepare for Direct Co-Disposal in a Renovated Reactor Building	Repackage and Prepare to Ship[b]	Melt and Dilute	Melt and Dilute in a Renovated Reactor Building	Mechanical Dilution	Vitrification Technologies	Electro-metallurgical Treatment	Conventional Processing[c]	Continued Wet Storage
Wet storage and handling	676	766	NA	676	766	676	676	676	655	1,650
Transfer, storage, and treatment	1,241	919	NA	1,363	1,073	1,566	2,411[d]	2,625	765	0
Fuel and waste processing	33	37	NA	47	55	46	67	67	610	78
Repository disposal[e]	169	169	NA	56	56	82	198[f]	23	36	0
Total	2,119	1,891	NA	2,142	1,950	2,370	3,352[g]	3,391	2,066[h]	1,730

EC

a. DOE (1998b).
b. Repackage and Prepare to Ship activities would be accomplished under Prepare for Direct Co-Disposal. The costs are included in those reported for Prepare for Co-Disposal. The material would be shipped to another DOE site. NA = not applicable.

L14-4

c. Value is for Conventional Processing of SNF until FY 2010, followed by Melt and Dilute treatment of later fuel receipts. This reduces the size of the facility for the later fuel receipts and lowers the overall life-cycle cost of the Conventional Processing technology by about $400 million.
d. Value is for the Dissolve and Vitrify technology. Glass Material Oxidation and Dissolution System would be 2,145 and Plasma Arc Treatment would be 2,143.
e. Costs for shipping the final waste form from SRS to a repository are included in these cost projections.
f. Value is for the Dissolve and Vitrify and Glass Material Oxidation and Dissolutions System technologies. Plasma Arc Treatment would be 80.
g. Value is for the Dissolve and Vitrify technology. Glass Material Oxidation and Dissolution System would be 3,087 and Plasma Arc Treatment would be 2,966.

L5-16, L14-3

h. Credits for sale of recovered enriched uranium are not included because of the recently signed agreement between Russia and the U.S. that calls for potential deferment of enriched uranium sales. Including these credits would decrease the cost of the electrometallurgical treatment by about $150 million and the Conventional Processing by about $110 million.

CHAPTER 3. AFFECTED ENVIRONMENT

This chapter describes the existing environmental and socioeconomic characteristics of the Savannah River Site (SRS) and the nearby region that the proposed action or its alternatives (described in Chapter 2) could affect. It provides the environmental bases against which the U.S. Department of Energy (DOE) has assessed the environmental consequences described in Chapter 4.

The activities that DOE describes in this environmental impact statement (EIS) would occur on the SRS, primarily in industrialized areas (for example see Figure 2-13). The only exception would involve the transportation of spent nuclear fuel or waste between SRS areas.

The industrialized areas consist primarily of buildings, paved parking lots, and graveled areas. There are grassed areas around some buildings, and there is vegetation along drainage ditches, but most of the industrialized areas have little or no vegetation.

As discussed in Section 2.4.2, DOE has identified three candidate host sites for the potential construction of a Transfer and Storage Facility. These sites are the east side of L Area inside the facility fence (see Figure 2-8), the southeast side of C Area inside the facility fence (see Figure 2-13), and the northeast side of P Area (see Figure 2-14). DOE also could construct a new Transfer, Storage and Treatment Facility at any of these three sites or in F or H Area. Finally, facilities to implement the New Processing Technology options could be located inside a reactor building, such as Building 105-L.

3.1 Geologic Setting and Seismicity

The SRS is in west-central South Carolina, approximately 100 miles from the Atlantic coast (Figure 3.1-1). It is on the Aiken Plateau of the Upper Atlantic Coastal Plain about 25 miles (40 kilometers) southeast of the Fall Line which separates the Atlantic Coastal Plain from the Piedmont.

3.1.1 GENERAL GEOLOGY

In South Carolina, the Atlantic Coastal Plain Province consists of a wedge of seaward-dipping and thickening unconsolidated and semiconsolidated sediments that extend from the Fall Line to the Continental Shelf (Figure 3.1-1). The Aiken Plateau is the subdivision of the Coastal Plain that includes the location of the SRS. The plateau extends from the Fall Line to the oldest of several scarps incised in the Coastal Plain sediment. The Plateau surface is highly dissected and characterized by broad interfluvial areas with narrow steep-sided valleys. It is generally well drained, although poorly drained depressions (called Carolina bays) occur (DOE 1995a). At the Site, the plateau is underlain by 500 to 1,400 feet (150 to 420 meters) of sands, clays, and limestones of Tertiary and Cretaceous age. These sediments are underlain, in turn, by sandstones of Triassic age and older metamorphic and igneous rocks (Arnett and Mamatey 1996). Because of the proximity of the SRS to the Piedmont Province, it has more relief than areas that are nearer the coast, with onsite elevations ranging from 89 to 420 feet (27 to 128 meters) above mean sea level.

The sediments of the Atlantic Coastal Plain (Figure 3.1-2) dip gently seaward from the Fall Line and range in age from Late Cretaceous to Recent. The sedimentary sequence thickens from essentially 0 at the Fall Line to more than 4,000 feet (1,219 meters) 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 (DOE 1995a). The formations of interest in C, F, H, L, and P Areas are part of the shallow (Floridan) aquifer system (Figure 3.1-2 and Table 3.1-1). Any contaminants could migrate to these formations and be carried by them to SRS streams.

Figure 3.1-1. General location of Savannah River Site and its relationship to physiographic provinces of southeastern United States.

Figure 3.1-2. Generalized geologic and aquifer units in SRS region.

Table 3.1-1. Soil formations of the Floridan aquifer system.[a]

Aquifer Unit	Formation	Description
Upper Three Runs Aquifer	"Upland Unit"	Poorly sorted, clayey-to-silty sands, with lenses and layers of conglomerates, pebbly sands, and clays. Clay clasts are abundant, and cross-bedding and flecks of weathered feldspar are locally common.
	Tobacco Road Formation	Moderately to poorly sorted, variably colored, fine-to-coarse grained sand, pebbly sand, and minor clay beds
	Dry Branch Formation	Variably colored, poorly sorted to well sorted sand with interbedded tan to gray clay
	Clinchfield Formation	Light colored basal quartz sand and glauconitic, biomoldic limestone, calcareous sand and clay. Sand beds of the formation constitute Riggins Mill Member and consist of medium to coarse, poorly to well sorted, loose and slightly indurated, tan, gray, and green quartz. The carbonate sequence of the Clinchfield consists of Utley Member -- sandy, glauconitic limestone and calcareous sand with indurated biomoldic facies
	Tinker/Santee Formation	Unconsolidated, moderately sorted, subangular, lower coarse-to-medium grained, slightly gravely, immature yellow and tan quartz sand and clayey sand; calcareous sands and clays and limestone also occur in F- and H-Areas.
Gordon Confining Unit (green clay)	Blue Bluff Member of Santee Limestone	Micritic limestone
	Warley Hill Formation	Fine grained, glauconitic, clayey sand, and clay that thicken, thin, and pinch out abruptly
Gordon Aquifer	Congaree Formation	Yellow, orange, tan, gray, and greenish gray, well-sorted, fine-to-coarse-grained quartz sands. Thin clay laminae occur throughout the section, with pebbly layers, clay clasts, and glauconite in places. In some places on SRS, upper part of Congaree Formation is cemented with silica; in other places it is slightly calcareous. Glauconitic clay, encountered in some borings on SRS near the base of this formation, indicates that basal contact is unconformable
	Fourmile Formation	Tan, yellow-orange, brown, and white, moderately to well-sorted sand, with clay beds near middle and top of unit. The sand is very coarse to fine-grained, with pebbly zones common. Glauconite and dinoflagellate fossils occur.
	Snapp Formation	Silty, medium- to course-grained quartz sand interbedded with clay. Dark, micaceous, lignitic sand also occurs. In northwestern part of SRS, this Formation is less silty and better sorted, with thinner clay interbeds.

a. Source: Aadland, Gellici, and Thayer (1995).

3.1.2 SUBSURFACE FEATURES

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There are several fault systems off the Site northwest of the Fall Line (DOE 1990a). A more recent study of geophysical evidence (Wike, Moore-Shedrow, and Shedrow 1996) and an earlier study (Stephenson and Stieve 1992) identified the faults indicated on Figure 3.1-3. The earlier study identified the following faults – Pen Branch, Steel Creek, Advanced Tactical Training Area (ATTA), Crackerneck, Ellenton, and Upper Three Runs – under SRS. The one closest to the areas under consideration is the Steel Creek Fault, which passes through L Area and is approximately 1 mile (1.6 kilometers) northwest of P Area. The Upper Three Runs Fault, which is a Paleozoic fault that does not cut Coastal Plain sediments, passes approximately 1 mile (1.6 kilometers) from F Area. The lines shown on Figure 3.1-3 represent the projection of faults to the ground surface. The actual faults do not reach the surface, but rather stop several hundred feet below.

Based on the available information, none of the faults discussed in this section is capable, which means that it has not moved at or near the ground surface within the past 35,000 years or is associated with another fault that had moved in the past 35,000 years. (10 CFR 100 contains a more detailed definition of a capable fault.)

3.1.3 SEISMICITY

Two major earthquakes have occurred within 186 miles (300 kilometers) of SRS.

- The Charleston, South Carolina, earthquake of 1886 had an estimated Richter scale magnitude of 6.8; it occurred approximately 90 miles (145 kilometers) from the SRS area, which experienced an estimated peak horizontal acceleration of 10 percent of gravity (0.10*g*) (URS/Blume 1982).

- The Union County, South Carolina, earthquake of 1913 had an estimated Richter scale magnitude of 6.0 and occurred about 99 miles (160 kilometers) from the Site (Bollinger 1973).

Because these earthquakes are not associated conclusively with a specific fault, researchers cannot determine the amount of displacement resulting from them.

In recent years, three earthquakes occurred inside the SRS boundary as reported by local print and media and cited in DOE (1999a). EC

- On May 17, 1997, with a Richter scale magnitude of 2.3 and a focal depth of 3.38 miles (5.44 kilometers); its epicenter was southeast of K Area.

- On August 5, 1988, with a local Richter scale magnitude of 2.0 and a focal depth of 1.66 miles (2.68 kilometers); its epicenter was northeast of K Area.

- On June 8, 1985, with a local Richter scale magnitude of 2.6 and a focal depth of 0.59 mile (0.96 kilometer); its epicenter was south of C Area and west of K Area.

Existing information does not relate these earthquakes conclusively with known faults under the Site. Figure 3.1-3 shows the locations of the epicenters of these earthquakes.

Outside the SRS boundary, an earthquake with a Richter scale magnitude of 3.2 occurred on August 8, 1993, approximately 10 miles (16 kilometers) east of the City of Aiken near Couchton, South Carolina. People reported feeling this earthquake in Aiken, New Ellenton (immediately north of SRS), North Augusta [approximately 25 miles (40 kilometers) northwest of the SRS], and on the Site (Aiken Standard 1993). EC

3.2 Water Resources

3.2.1 SURFACE WATER RESOURCES

This section describes the surface water, and the quality of that water, in the area potentially af-

fected by the proposed action, including the Savannah River, Upper Three Runs, Fourmile Branch, and Steel Creek.

Figure 3.1-3. Savannah River Site, showing seismic fault lines and locations of onsite earthquakes and their year of occurrence.

3.2.1.1 Savannah River

The Savannah River bounds SRS on its southwestern border for about 20 miles (32 kilometers), approximately 160 river miles (260 river kilometers) from the Atlantic Ocean. 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. River flow averages about 10,000 cubic feet (283 cubic meters) per second at SRS (DOE 1995a).

The Savannah River, which forms the boundary between Georgia and South Carolina, supplies potable water to a number of users. Upstream of SRS, the river supplies domestic and industrial water for Augusta, Georgia, and North Augusta, South Carolina. Approximately 130 river miles (210 river kilometers) downstream of SRS, the river supplies domestic and industrial water for Savannah, Georgia, and Beaufort and Jasper Counties in South Carolina through intakes at about River Mile 29 and River Mile 39, respectively (DOE 1995b).

The Savannah River receives sewage treatment plant effluent from Augusta, Georgia; North Augusta, Aiken, and Horse Creek Valley, South Carolina; and from a number of SRS operations through discharges to onsite streams. In addition, the Georgia Power Company's Vogtle Electric Generating Plant withdraws an average of 46 cubic feet (1.3 cubic meters) per second for cooling and returns an average of 12 cubic feet (0.35 cubic meter) per second of cooling tower blowdown. The Urquhart Steam Generating Station at Beech Island, South Carolina, withdraws approximately 265 cubic feet (7.5 cubic meters) per second for once-through cooling water (DOE 1995a).

On SRS, a swamp occupies the floodplain along the Savannah River for approximately 10 miles (17 kilometers); the swamp is about 1.5 miles (2.5 kilometers) wide. A natural levee separates the river from the floodplain. Figure 3.2-1 shows the 100-year floodplain of the Savannah River in the SRS vicinity and the floodplains of major tributaries that drain the Site (DOE 1995a).

3.2.1.2 SRS Streams

Five tributaries of the Savannah River -- Upper Three Runs, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs -- drain almost all of the SRS (Figure 3.2-1). Each stream 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 received varying amounts of effluent from SRS operations, are not commercial sources of water. Their natural flows range from less than 10 cubic feet (l 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 (DOE 1995a).

Upper Three Runs, Fourmile Branch, and Steel Creek are the streams closest to most SRS spent nuclear fuel management locations (see Figure 3.2-1). These streams also are closest to the areas where DOE is most likely to place new spent nuclear fuel facilities.

Upper Three Runs is a large, cool, blackwater stream in the northern part of 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 its mouth. Upper Three Runs is approximately 25 miles (40 kilometers) long, with its lower 17 miles (28 kilometers) inside SRS boundaries. This creek receives more water from underground sources than other SRS streams and, therefore, has lower conductivity, hardness, and pH values. Upper Three Runs is the only major tributary on SRS that has never received thermal discharges from nuclear reactors (DOE 1995a).

Fourmile Branch is about 15 miles (24 kilometers) long and drains an area of approximately 22 square miles (57 square kilometers). At its headwaters, Fourmile Branch is a small blackwater stream that currently receives

Figure 3.2-1. Savannah River Site, showing 100-year floodplain and major stream systems.

impacts from SRS operations. The water chemistry in the headwater area is very similar to that of Upper Three Runs, with the exception of nitrate concentrations, which are an order of magnitude higher than those in Upper Three Runs (DOE 1995a). These elevated 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 discharges primarily into the Savannah River at River Mile 152, while a small portion flows west and enters Beaver Dam Creek, a small onsite tributary of the Savannah River (DOE 1995a). EC

Steel Creek is about 9 miles (15 kilometers) long and, with Meyers Branch, drains an area of approximately 35 square miles (90 square kilometers) (DOE 1996a). Its headwaters originate near P Reactor. The creek flows southwest about 2 miles (3 kilometers) before it enters the headwaters of L Lake. Flow from the outfall of the L-Lake dam travels about 3 miles (5 kilometers) before entering the Savannah River swamp and then another 2 miles (3 kilometers) before entering the river.

Meyers Branch, the main tributary of Steel Creek, flows approximately 6 miles (10 kilometers) before entering Steel Creek. Meyers Branch is a small blackwater stream that has remained relatively undisturbed by SRS operations. The confluence of Meyers Branch and Steel Creek is downstream from the L-Lake dam. Steel Creek received intermittent thermal effluent from P and L Reactors from 1954 to 1964, and from L Reactor only from 1964 to 1968 (Halverson et al. 1997). Effluents from L and P Areas flow to L Lake and subsequently to Steel Creek through the L-Lake dam outfall. During water year 1996, flows in Steel Creek (downstream of the confluence with Meyers Branch) averaged 59.2 cubic feet (1.7 cubic meters) per second (DOE 1996a).

3.2.1.3 Surface-Water Quality

In 1996, releases of radionuclides from the SRS to surface waters amounted to 8,550 curies of tritium, 0.214 curie of strontium-89 and -90, and 0.05 curie of plutonium-239 (Arnett and Mamatey 1998a). Table 3.2-1 lists radioactive liquid releases by source for 1997; Table 3.2-2 lists radioactive liquid releases by outfall or facility and compares annual average radionuclide concentrations to DOE concentration guides (Figure 3.2-2 shows outfall and facility locations for radioactive surveillance). The resulting doses to a downriver consumer of river water from radionuclides released from the Site were less than 2 percent of the U.S. Environmental Protection Agency (EPA) and DOE standards for public water supplies (40 CFR Part 141 and DOE Order 5400.5, respectively) and less than 0.2 percent of the DOE dose standard from all pathways (DOE 1990b; Arnett and Mamatey 1998). TC TC

The South Carolina Department of Health and Environmental Control (SCDHEC) regulates the physical properties and concentrations of chemicals and metals in SRS effluents under the National Pollutant Discharge Elimination System (NPDES) program. SCDHEC, which also regulates biological water quality standards for SRS waters, has classified the Savannah River and SRS streams as “Freshwaters.” In 1997, 99.9 percent of the NPDES water quality analyses on SRS effluents were in compliance with the SRS NPDES permit; only 7 of 5,758 analyses exceeded permit limits (Arnett and Mamatey 1998a). A comparison of 1997 Savannah River water quality analysis upstream and downstream of SRS showed no significant differences, and a comparison with historical data indicates that coliform data are within normal fluctuation for river water in this area and the overall exceedances decreased in number from 1996 (Arnett and Mamatey 1998a). Table 3.2-3 summarizes the water quality of Fourmile Creek, Steel Creek, and Upper Three Runs for 1996. TC TC

Table 3.2-1. Annual liquid releases by source for 1997 (including direct and seepage basin migration releases).[a]

		Curies					
Radionuclide[b]	Half-life (years)	Reactors	Separations[c]	Reactor materials	TNX	SRTC	Total
H-3 (oxide)	12.3	2.91×10^{3}	5.24×10^{-3}		4.02×10^{2}	1.82	8.55×10^{-3}
Sr-89,90[d]	29.1	6.46×10^{-2}	1.40×10^{-1}		5.09×10^{-3}	4.10×10^{-3}	2.14×10^{-1}
I-129[e]	1.6×10^{7}		7.82×10^{-2}[e]				7.82×10^{-2}[d]
Cs-137	30.2	2.86×10^{-3}	4.49×10^{-2}				4.78×10^{-2}
U-234	2.46×10^{5}	4.45×10^{-3}	2.30×10^{-2}	2.68×10^{-5}	1.52×10^{-6}	1.06×10^{-4}	2.76×10^{-2}
U-235	7.04×10^{8}	4.91×10^{-5}	7.23×10^{-4}		1.37×10^{-7}	3.44×10^{-6}	7.76×10^{-4}
U-238	4.47×10^{9}	3.83×10^{-3}	2.57×10^{-2}	5.71×10^{-5}	9.19×10^{-6}	1.11×10^{-4}	2.97×10^{-2}
Pu-238	87.7	4.24×10^{-5}	9.57×10^{-4}		7.68×10^{-7}	1.78×10^{-6}	1.00×10^{-3}
Pu-239[f]	24,100	1.10×10^{-2}	3.39×10^{-2}	1.14×10^{-3}	1.12×10^{-3}	3.38×10^{-3}	5.05×10^{-2}
Am-241	432.7		7.81×10^{-6}	2.11×10^{-6}			9.92×10^{-6}
Cm-244	18.1		2.93×10^{-6}	4.14×10^{-7}			3.34×10^{-6}

Notes: Blank spaces indicate no quantifiable activity.
a. Source: Arnett and Mamatey (1998a).
b. H = hydrogen (H-3 = tritium), Sr = strontium, I = iodine, Cs = cesium, U = uranium, Pu = plutonium, Am = americium, Cm = curium.
c. Includes separations, waste management, and tritium facilities.
d. Includes unidentified beta.
e. Measured I-129 doses were not available for 1997. The value for separations emissions is from 1996.
f. Includes unidentified alpha.
TNX = a technology development facility adjacent to the Savannah River.
SRTC = Savannah River Technology Center.

TC

(Figure 3.2-3 shows stream water quality monitoring locations.)

Certain technologies, including those considered in this EIS, generate liquid byproducts that are transferred to the F- and H-Area Tank Farms. Evaporator overheads from these tanks are condensed and treated at the F- and H-Area Effluent Treatment Facility (ETF). Waste concentrate from the ETF is disposed of in the Z-Area Saltstone Manufacturing and Disposal Facility and the decontaminated wastewater is discharged to Upper Three Runs through NPDES outfall H-16. These existing facilities are described in the *Interim Management of Nuclear Materials EIS* (DOE 1995b) and the *Defense Waste Processing Facility Supplemental EIS* (DOE 1994). Requirements for spent nuclear fuel processing are included in these documents and, therefore, this EIS considers those facilities and processed waste amounts to be part of the SRS baseline.

3.2.2 GROUNDWATER RESOURCES

3.2.2.1 Groundwater Features

In the SRS region, the subsurface contains two hydrogeologic provinces. The uppermost, consisting of a wedge of unconsolidated Coastal Plain sediments of Late Cretaceous and Tertiary age, is the Atlantic Coastal Plain Hydrogeologic Province. Beneath the sediments of the Atlantic Coastal Plain Hydrogeologic Province are rocks of the Piedmont Hydrogeologic Province. These rocks consist of Paleozoic igneous and metamorphic basement rocks and lithified mudstone, sandstone, and conglomerates of the Dunbarton basin of the Upper Triassic. Sediments of the Atlantic Coastal Plain Hydrogeologic Province are divided into three main

Table 3.2-2. Liquid radioactive releases by outfall/facility and comparison of annual average radionuclide concentrations to DOE derived concentration guides.[a]

Outfall or Facility	Radionuclide[b]	Quantity of Radionuclides Released during 1997 (Ci)	Average Effluent Concentration during 1997 (μCi/mL)	DOE DCGs[c] (μCi/mL)
C Area (C Reactor)				
C Canal	H-3 (oxide)	1.20	1.75×10^{-6}	2.00×10^{-3}
	Sr-89,90	Below MDL	ND	1.00×10^{-6}
	Cs-137	Below MDL	1.02×10^{-9}	3.00×10^{-6}
F Area (Separations and Waste Management)				
F-01	H-3 (oxide)	5.03×10^{-2}	2.54×10^{-7}	2.00×10^{-3}
	Sr-89,90	Below MDL	ND	1.00×10^{-6}
	Cs-137	Below MDL	1.32×10^{-9}	3.00×10^{-6}
F-012 (281-8F Retention Basin)	H-3 (oxide)	7.67×10^{-1}	9.83×10^{-6}	2.00×10^{-3}
	Sr-89,90	Below MDL	3.01×10^{-9}	1.00×10^{-6}
	Cs-137	158×10^{-3}	2.07×10^{-8}	3.00×10^{-6}
	H-3 (oxide)	1.73×10^{-2}	1.63×10^{-6}	2.00×10^{-3}
	Sr-89,90	3.13×10^{-5}	4.39×10^{-9}	1.00×10^{-6}
	Cs-137	5.92×10^{-4}	2.30×10^{-8}	3.00×10^{-6}
Fourmile Branch-3 (F-Area Effluent)	H-3 (oxide)	1.32×10	7.80×10^{-7}	2.00×10^{-3}
	Sr-89,90	Below MDL	4.16×10^{-10}	1.00×10^{-6}
	Cs-137	Below MDL	8.97×10^{-10}	3.00×10^{-6}
Upper Three Runs-2 (F Storm Sewer)	H-3 (oxide)	1.66×10^{-1}	8.78×10^{-7}	2.00×10^{-3}
	Sr-89,90	Below MDL	8.56×10^{-11}	1.00×10^{-6}
	Cs-137	Below MDL	5.13×10^{-10}	3.00×10^{-6}
	U-234	6.86×10^{-5}	3.48×10^{-10}	6.00×10^{-7}
	U-235	5.15×10^{-6}	3.02×10^{-11}	6.00×10^{-7}
	U-238	1.90×10^{-4}	9.15×10^{-10}	6.00×10^{-7}
	Pu-238	1.54×10^{-5}	9.10×10^{-11}	4.00×10^{-8}
	Pu-239	7.73×10^{-6}	4.66×10^{-11}	3.00×10^{-8}
	Am-241	7.77×10^{-6}	3.98×10^{-11}	3.00×10^{-8}
	Cm-244	2.92×10^{-6}	1.74×10^{-11}	6.00×10^{-8}
Upper Three Runs F-3 (Naval Fuel Effluent)	H-3 (oxide)	3.45×10^{-2}	1.46×10^{-6}	2.00×10^{-3}
	Sr-89,90	Below MDL	1.16×10^{-10}	1.00×10^{-6}
	Cs-137	Below MDL	2.47×10^{-10}	3.00×10^{-6}
	U-234	1.62×10^{-5}	8.95×10^{-10}	6.00×10^{-7}
	U-235	5.86×10^{-6}	2.30×10^{-9}	6.00×10^{-7}
	U-238	3.04×10^{-6}	1.76×10^{-10}	6.00×10^{-7}
	Pu-238	1.61×10^{-7}	6.23×10^{-12}	4.00×10^{-8}
	Pu-239	2.60×10^{-8}	5.04×10^{-12}	3.00×10^{-8}
	Am-241	4.49×10^{-8}	7.07×10^{-13}	3.00×10^{-8}
	Cm-244	9.54×10^{-9}	-6.84×10^{-11}	6.00×10^{-8}

TC

Table 3.2-2. (continued).

Outfall or Facility	Radionuclide[b]	Quantity of Radionuclides Released during 1997 (Ci)	Average Effluent Concentration during 1997 (µCi/mL)	DOE DCGs[c] (µCi/mL)
H Area (Separations and Waste Management)				
Fourmile Branch-1C (H-Area Effluent)	H-3 (oxide)	3.85×10	9.22×10^{-6}	2.00×10^{-3}
	Sr-89,90	7.93×10^{-5}	7.05×10^{-10}	1.00×10^{-6}
	Cs-137	6.77×10^{-4}	3.27×10^{-9}	3.00×10^{-6}
	H-3 (oxide)	4.96×10^{-1}	1.23×10^{-5}	2.00×10^{-3}
	Sr-89,90	3.48×10^{-6}	5.40×10^{-10}	1.00×10^{-6}
	Cs-137	2.15×10^{-6}	7.15×10^{-10}	3.00×10^{-6}
	U-234	2.77×10^{-6}	8.54×10^{-11}	6.00×10^{-7}
	U-235	9.84×10^{-9}	8.61×10^{-12}	6.00×10^{-7}
	U-238	2.07×10^{-6}	6.58×10^{-11}	6.00×10^{-7}
	Pu-238	5.09×10^{-7}	2.45×10^{-11}	4.00×10^{-8}
	Pu-239	8.93×10^{-8}	6.37×10^{-12}	3.00×10^{-8}
H-017 (281-8H Retention Basin)	H-3	7.17×10^{-1}	1.02×10^{-5}	2.00×10^{-3}
	Sr-89,90	5.21×10^{-4}	7.91×10^{-9}	1.00×10^{-6}
	Cs-137	1.04×10^{-2}	1.11×10^{-7}	3.00×10^{-6}
H-018 (200-H Cooling Basin)	H-3 (oxide)	1.44×10^{-1}	2.27×10^{-5}	2.00×10^{-3}
	Sr-89,90	2.75×10^{-4}	4.58×10^{-8}	1.00×10^{-6}
	Cs-137	2.21×10^{-3}	3.71×10^{-7}	3.00×10^{-6}
HP-15 (Tritium Facility Outfall)	H-3 (oxide)	1.74×10	1.55×10^{-5}	2.00×10^{-3}
	Cs-137	Below MDL	7.75×10^{-11}	3.00×10^{-6}
HP-52 (H-Area Tank Farm)	H-3 (oxide)	2.43×10	1.30×10^{-6}	2.00×10^{-3}
	SR-89,90	Below MDL	7.67×10^{-11}	1.00×10^{-6}
	Cs-137	1.58×10^{-4}	1.92×10^{-9}	3.00×10^{-6}
TC McQueen's Branch at Rd F	H-3 (oxide)	120×10^{1}	1.05×10^{-5}	2.00×10^{-3}
	Cs-137	Below MDL	4.85×10^{-10}	3.00×10^{-6}
Upper Three Runs – 2A (ETF[e] Outfall at Rd C)	H-3 (oxide)	3.82×10^{2}	(f)	2.00×10^{-3}
	Sr-89,90	1.28×10^{-5}	2.24×10^{-9}	1.00×10^{-6}
	Cs-137	1.79×10^{-2}	2.16×10^{-7}	3.00×10^{-6}
L Area (L Reactor)				
L-007	H-3 (oxide)	6.02×10	3.38×10^{-7}	2.00×10^{-3}
	Sr-89,90	Below MDL	1.16×10^{-10}	1.00×10^{-6}
	Cs-137	Below MDL	4.53×10^{-10}	3.00×10^{-6}
P Area (P Reactor)				
P-013A	H-3 (oxide)	7.18×10^{-1}	2.96×10^{-4}	2.00×10^{-3}
	Sr-89,90	5.25×10^{-6}	3.47×10^{-9}	1.00×10^{-6}
	Cs-137	2.38×10^{-4}	9.86×10^{-8}	3.00×10^{-6}
P-019A (P-Area Canal Par Pond)	H-3 (oxide)	3.25×10^{-1}	5.41×10^{-7}	2.00×10^{-3}
	Sr-89,90	Below MDL	3.03×10^{-10}	1.00×10^{-6}
	Cs-137	Below MDL	ND	3.00×10^{-6}

a. Source: Arnett and Mamatey (1998a).
b. H = hydrogen (H-3 = tritium), Sr = strontium, I = iodine, Cs = cesium, U = uranium, Pu = plutonium, Am = americium, Cm = curium.
c. DCG = derived concentration guide. Source: DOE Order 5400.5. In cases where different chemical forms have different DCGs, the lowest DCG for the radionuclide is given. DCGs are defined as the concentration of that radionuclide that will give a 50-year committed effective dose equivalent of 100 mrem under conditions of continuous exposure for one year. DCGs are reference values only and are not considered release limits or standards.
d. MDL = minimum detectable level.
e. ETF = Effluent Treatment Facility.
f. Outfall concentrations for tritium exceed the DCG guidelines. DOE Order 5400.5 exempts tritium from "best available technology" requirements because there is no practical technology available for removing tritium from dilute liquid waste streams.

ND = not detected.

Figure 3.2-2. Radiological surface-water sampling locations.

Table 3.2-3. SRS stream water quality (onsite downstream locations).[a]

Parameter[b]	Units	Fourmile Branch (FM-6) average	Steel Creek (SC-4) average	Upper Three Runs (U3R-4) average	Water Quality Criterion[c], MCL[d], or DCG[e]
Aluminum	Mg/L	0.200[f]	0.018	0.274[f]	0.087
Cadmium	Mg/L	ND[g]	ND	ND	0.00066
Calcium	Mg/L	2.94	2.53	1.62	NA[h]
Cesium-137	PCi/L	NR[i]	NR	NR	120[e]
Chromium	mg/L	ND	ND	ND	0.011
Copper	mg/L	0.015[f]	0.028[f]	0.036[f]	0.0065
Dissolved oxygen	mg/L	7.9	8.73	8.2	≥5
Iron	mg/L	0.69	0.349	0.586	1
Lead	mg/L	ND	ND	ND	0.0013
Magnesium	mg/L	0.659[f]	0.854[f]	0.385[f]	0.3
Manganese	mg/L	0.055	0.048	0.026	1
Mercury	mg/L	ND	0.0002	ND	0.000012
Nickel	mg/L	0.01	0.01	0.012	0.088
Nitrate (as nitrogen)	mg/L	1.36	0.16	0.24	10[d]
pH	pH	6.31	6.32	6.3	6-8.5
Plutonium-238	pCi/L	NR	NR	NR	1.6[e]
Plutonium-239	pCi/L	NR	NR	NR	1.2[e]
Sodium	mg/L	6.8	1.89	1.58	NA
Strontium-89,90	pCi/L	NR	NR	NR	8[d]
Suspended solids	mg/L	8.08	5.2	14.1	NA
Temperature[j]	°C	18.1	18.6	17.3	32.2
Total dissolved solids	mg/L	355.6	48	36	500[k]
Tritium	pCi/L	NR	NR	NR	20,000[d]
Uranium-234	pCi/L	NR	NR	NR	20[e]
Uranium-235	pCi/L	NR	NR	NR	24[e]
Uranium-238	pCi/L	NR	NR	NR	24[e]
Zinc	mg/L	0.041	0.040	0.028	0.059

a. Source: Arnett and Mamatey (1997).
b. Parameters DOE routinely measures as a regulatory requirement or as part of ongoing monitoring programs.
c. Water Quality Criterion (WQC) is Aquatic Chronic Toxicity unless otherwise indicated.
d. MCL = Maximum Contaminant Level; State Primary Drinking Water Regulations.
e. DCG = DOE Derived Concentration Guides for Water (DOE Order 5400.5). 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, value listed is 4 percent of DCG.
f. Concentration exceeded WQC; however, these criteria are for comparison only. WQCs are not legally enforceable.
g. ND = Not Detected.
h. NA = Not Applicable.
i. NR = Not Reported.
j. Shall not be increased more than 2.8°C (5°F) above natural temperature conditions or exceed a maximum of 32.2°C (90°F) as a result of the discharge of heated liquids unless appropriate temperature criterion mixing zone has been established.
k. Secondary MCL; State Primary Drinking Water Regulations.

Figure 3.2-3. SRS streams and Savannah River water quality sampling locations.

aquifer systems, the Floridan Aquifer System, the Dublin Aquifer System, and the Midville Aquifer System as shown in Figure 3.1-2 (Aadland, Gellici, and Thayer 1995). Each aquifer system is divided from the others by two confining systems, the Meyers Branch Confining System and the Allendale Confining System, as shown in Figure 3.1-2.

Groundwater within the Floridan system (the shallow aquifer beneath the Site) flows slowly toward SRS streams and swamps and into the Savannah River at rates ranging from inches to several hundred feet per year. The depth to which onsite streams cut into soils and the orientation of the soil formations control the horizontal and vertical movement of the groundwater. The valleys of smaller perennial streams allow discharge from the shallow saturated geologic formations. The valleys of major tributaries of the Savannah River (e.g., Upper Three Runs) drain formations of intermediate depth, and the river valley drains deep formations. With the release of water to the streams, the hydraulic head of the aquifer unit releasing the water can become less than that of the underlying unit. If this occurs, groundwater has the potential to migrate from the lower unit to the overlying unit.

Groundwater flow in the shallow aquifer (Floridan) system is vertically downward in the divide areas between surface water drainages due to the decreasing hydraulic head with increasing depth. In areas along the lower reaches of most of the Site streams, groundwater moves vertically upward from deeper aquifers to the shallow aquifers. In these areas hydraulic heads increases with depth.

In the vicinity of these streams, the vertical upward flow occurs across the Crouch Branch Confining Unit/Gordon Confining Unit. At these locations any contaminants in the overlying aquifer system are prevented from migrating into deeper aquifers by the prevailing hydraulic gradient and the low permeability of the confining unit. Horizontal groundwater flow occurs at the M-Area metallurgical laboratory (to the west-northwest in the shallow aquifer and subsequent flow to the south toward Upper Three Runs in the intermediate aquifer), K-Area Disassembly Basin (toward Pen Branch and L Lake), P-Area Disassembly Basin (toward Steel Creek), F Canyon (toward Upper Three Runs and Fourmile Branch), and H Canyon (toward Upper Three Runs and its tributaries).

3.2.2.2 Groundwater Use

Groundwater is a domestic, municipal, and industrial water source throughout the Upper Coastal Plain. Domestic water supplies come primarily from the shallow aquifers including the Gordon Aquifer and the Upper Three Runs Aquifer (water-table aquifer). Most municipal and industrial water supplies in Aiken County are from the Cretaceous intermediate to deep aquifer units. In Barnwell and Allendale Counties some municipal water supplies are from the Gordon Aquifer and overlying units that thicken to the southeast. At SRS, most groundwater production for domestic and process water comes from the intermediate/deep aquifers (i.e., the Crouch Branch and McQueen Branch Aquifers), with a few lower-capacity process water wells pumping from the shallower Gordon Aquifer.

Every major operating area at SRS has groundwater wells; total groundwater production ranges from 9 to 12 million gallons (34,000 to 45,000 cubic meters) per day, similar to the volume pumped for industrial and municipal production within 10 miles (16 kilometers) of the Site (Arnett and Mamatey 1996).

From October 1995 to September 1996, the total groundwater withdrawal rate for C, F, H, P, and L Areas was approximately 4 million gallons (15,130 cubic meters) per day. Groundwater in C Area comes from two domestic wells that produced approximately 220,000 gallons (830 cubic meters) per day. Groundwater in F Area is pumped from four process production and two domestic wells. The total F-Area groundwater production rate from October 1995 to September 1996 was approximately 1.58 million gallons (5,981 cubic meters) per day. During the same period, wells in H, L, and P Areas produced ap-

proximately 1.9 million gallons (7,190 cubic meters) per day, 140,000 gallons (530 cubic meters) per day, and 170,000 gallons (640 cubic meters) per day, respectively. H Area has two domestic wells and three process production wells; L Area has two domestic wells. Until recently, two P-Area groundwater wells were used for domestic purpose. At present, these wells are not being used for domestic or process production. SRS is implementing a consolidation program for domestic wells. When this program is complete, DOE might take the domestic wells in C, F, H, and L Areas out of service or use them only for process water (Wells 1997).

3.2.2.3 SRS Hydrogeology

The aquifers of interest for C, F, H, L, and P Areas are the Upper Three Runs and Gordon Aquifers. The Upper Three Runs (water table) Aquifer is defined by the hydrogeologic properties of the Tinker/Santee Formation, the Dry Branch Formation, and the Tobacco Road Formation (DOE 1996a). Table 3.1-1 lists these formations.

The Gordon Confining Unit (green clay), which separates the Upper Three Runs and Gordon Aquifers, consists of the Warley Hill Formation and the Blue Bluff Member of the Santee Limestone (Table 3.1-1). It is not a continuous clay unit, but consists of several lenses of green and gray clay that thicken, thin, and pinch out abruptly. Locally, beds of calcareous mud add to the thickness of the unit with minor interbeds of clayey sand or sand. The vertical hydraulic conductivity ranges from 1.1×10^{-6} foot (3.4×10^{-5} centimeter) to 0.16 foot (4.9 centimeters) per day and the horizontal conductivity ranges from 5.4×10^{-6} foot (1.6×10^{-5} centimeter) to 5.7×10^{-3} foot (0.17 centimeter) per day (Aadland, Gellici, and Thayer 1995). TC

The Gordon Aquifer consists of the Congaree, Fourmile, and Snapp Formations. Table 3.1-1 lists the soil descriptions for these formations. The Gordon Aquifer is partially eroded near the Savannah River and Upper Three Runs. This aquifer is recharged directly by precipitation in the outcrop area and at interstream drainage divides in and near the outcrop area, and by leakage from overlying and underlying aquifers. The northeast-to-southwest hydraulic gradient across SRS is consistent and averages 4.8 feet per mile (0.9 meter per kilometer). Based on pumping tests on 13 SRS wells, the average hydraulic conductivity is approximately 35 feet (10.7 meters) per day.

3.2.2.4 Groundwater Quality

Industrial solvents, metals, tritium, and other constituents used or generated on SRS have contaminated the shallow aquifers beneath 5 to 10 percent of the Site. In general, DOE does not use these aquifers for SRS operations or drinking water, although there are a few low-yield wells in the Gordon Aquifer. The shallow aquifer units discharge to SRS streams and eventually the Savannah River (Arnett and Mamatey 1997). TC

Most contaminated groundwater at SRS occurs beneath a few facilities; the contaminants reflect the operations and chemical processes performed at those facilities. At C Area, groundwater contaminants above regulatory or SRS guidelines include tritium and other radionuclides, bis (2-ethylhexyl) phthalate, carbon disulfide, lead, manganese, and chlorinated organics. At F and H Areas, contaminants above the guidelines include tritium and other radionuclides, metals, nitrates, sulfates, and chlorinated and volatile organics. At L Area, tritium, other radionuclides, carbon disulfide, chlorinated and volatile organics, and metals are in the groundwater at levels above the guidelines. Groundwater beneath the L-Area Disassembly Basin has been affected by metals, chlorinated organics, and tritium at levels above regulatory guidelines. Tables 3.2-4 through 3.2-8 list concentrations of individual analytes above regulatory or SRS guidelines for 1995 in C, F, H, L, and P Areas, respectively (WSRC 1995a). Figure 3.2-4 shows generalized groundwater contamination maximum values for analytes at or above regulatory or established SRS guidelines for the areas of concern.

Table 3.2-4. C-Area maximum reported groundwater parameters in excess of regulatory and SRS limits.[a]

Analyte	Concentration	Regulatory Limit
Aluminum[b]	6,430 µg/L	50 µg/L[c]
Bis (2-ethylhexyl) phthalate	23 µg/L	6 µg/L[d]
Iron[b]	10,500 µg/L	300 µg/L[d]
Lead[b]	301 µg/L	50 µg/L[e]
Manganese[b]	254 µg/L	50 µg/L[c]
Carbon disulfide	74 µg/L	10 µg/L[f]
Trichloroethylene	1,580 µg/L	5 µg/L[d]
Tetrachloroethylene	174 µg/L	5 µg/L[d]
Dichloromethane	8.7 µg/L	5 µg/L[d]
Total organic halogens	972 µg/L	50 µg/L[f]
Tritium	2.4×10^{-2} µCi/mL	2.0×10^{-5} µCi/mL[d]
Thallium	3.5 µg/L	2 µg/L[d]
Thorium-234	6.8×10^{-7} µCi/mL	4.01×10^{-7} µCi/mL[g]

a. µg/L = micrograms per liter; µCi/mL = microcuries per milliliter.
b. Total recoverable.
c. EPA National Secondary Drinking Water Standards (WSRC 1995a).
d. EPA Primary Drinking Water Standards (WSRC 1995a).
e. SCDHEC Final Primary Drinking Water Standards (WSRC 1995a).
f. Drinking Water Standards do not apply. Criterion 10 times a recently published 90th percentile detection limit was used (WSRC 1995a).
g. EPA Proposed Primary Drinking Water Standard (WSRC 1995a).

3.3 Air Resources

3.3.1 GENERAL METEOROLOGY

Based on data collected from SRS meteorological towers from 1987 through 1991 (the latest quality-assured 5-year data set), maximum wind direction frequencies at the Site are from the northeast and west-southwest and the average wind speed is 8.5 miles per hour (3.8 meters per second). The average annual temperature at the Site is 64°F (17.8°C). The atmosphere in the region is unstable approximately 56 percent of the time, neutral 23 percent of the time, and stable about 21 percent of the time (Shedrow 1993). In general, as the atmosphere becomes more unstable, atmospheric dispersion of airborne pollutants increases and ground-level pollutant concentrations decrease.

3.3.2 SEVERE WEATHER

The SRS area experiences an average of 55 thunderstorm days a year, 50 percent of which occur in June, July, and August (Shedrow 1993). On average, lightning strikes six times a year on a square-kilometer area (Hunter 1990). The highest windspeed recorded at Bush Field (Augusta, Georgia) between 1950 and 1993 was 62 miles (100 kilometers) per hour (NOAA 1994).

From 1954 to 1983, 37 reported tornadoes occurred in a 1-degree square of latitude and longitude that includes SRS (WSRC 1993). This frequency of occurrence is equivalent to an average of about one tornado per year. Tornado statistics indicate that the average frequency of a tornado striking any single point on the site is TC

Table 3.2-5. F-Area maximum reported groundwater parameters in excess of regulatory and SRS limits.[a]

Analyte	Concentration (µg/L for metals and organics; µCi/mL for radioisotopes unless otherwise noted)	Regulatory limit (µg/L for metals and organics; µCi/mL for radioisotopes)
Aluminum[b]	95,900	50[c]
Beryllium[b]	10	4[d]
Bis (2-ethylhexyl) phthalate	190	6[d]
Cadmium[b]	243	5[d]
Copper[b]	1,210	1,000[d]
Chromium[b]	185	100[d]
Iron[b]	261,000	300[d]
Lead[b]	6,500	50[e]
Lithium[b]	249	50[f]
Manganese[b]	15,000	50[c]
Mercury[b]	5.4	2[e]
Nickel[b]	176	100[d]
Carbon tetrachloride	23	5[d]
Trichloroethylene	96	5[d]
Trichlorofluoromethane	80	10[f]
Tetrachloroethylene	42	5[d]
Dichloromethane	65	5[d]
1,2-dichloroethane	162	5[d]
Total organic carbon	18,600	10,000
Total organic halogens	148	50[f]
Nitrate as nitrogen	71,300	1,000[d]
Nitrate-nitrite as nitrogen	384,000	10,000[d]
Americium-241	9.9×10^{-8}	6.34×10^{-9g}
Cesium-137	4.4×10^{-7}	2.0×10^{-7h}
Cobalt[b]	665	40[f]
Curium-243/244	1.6×10^{-7}	8.3×10^{-9g}
Curium-245/246	9.9×10^{-8}	6.23×10^{-9g}
Iodine-129	7.2×10^{-7}	1.0×10^{-9h}
Lithium[b]	56	50[f]
Tritium	2.2×10^{-2}	2.0×10^{-5d}
Plutonium-238	2.3×10^{-8}	7.02×10^{-9g}
Radium-226	1.1×10^{-7}	$2.0\times10^{-8g,I}$
Radium-228	3.1×10^{-7}	$2.0\times10^{-8g,I}$
Nonvolatile beta	2.5×10^{-5}	5.0×10^{-8h}
Total alpha-emitting radium	1.6×10^{-7}	2.0×10^{-8g}
Gross alpha	2.5×10^{-6}	1.5×10^{-8d}
Strontium-89	7.1×10^{-7}	2.0×10^{-8h}
Strontium-90	7.4×10^{-6}	8.0×10^{-9d}
Thallium[b]	4.3	2.0[d]
Thorium-234	9.5×10^{-7}	4.01×10^{-7g}
Uranium-233/234	4.8×10^{-7}	1.38×10^{-8g}
Uranium-235	5.0×10^{-8}	1.45×10^{-8g}
Uranium-238	1.3×10^{-6}	1.46×10^{-8g}

a. Abbreviations: µg/L = micrograms per liter; µCi/mL = microcuries per milliliter.
b. Total recoverable.
c. EPA National Secondary Drinking Water Standard (WSRC 1995a).
d. EPA Final Primary Drinking Water Standard (WSRC 1995a).
e. SCDHEC Final Primary Drinking Water Standard (WSRC 1995a).
f. Drinking Water Standards do not apply. Criterion 10 times a recently published 90th percentile detection limit was used (WSRC 1995a).
g. EPA Proposed Primary Drinking Water Standard (WSRC 1995a).
h. EPA Interim Final Primary Drinking Water Standards (WSRC 1995a).
i. Radium-226/228 combined proposed Maximum Contaminant Level of 5.0×10^{-8} microcuries per milliliter.

Table 3.2-6. H-Area maximum reported groundwater parameters in excess of regulatory and SRS limits.[a]

Analyte	Concentration (µg/L for metals and organics; µCi/mL for radioisotopes)	Regulatory limit (µg/L for metals and organics; µCi/mL for radioisotopes)
Aluminum[b]	2,800	50[c]
Bis (2-ethylhexyl) phthalate	23	6[d]
Iron[b]	7,990	300[d]
Lead[b]	301	50[e]
Manganese[b]	91	50[c]
Trichloroethylene	1,580	50[c]
Total Organic Halogens	972	50[d]
Thallium[b]	4.0	2.0[d]
Tritium	2.4×10^{-2}	2.0×10^{-5}[d]
Thorium-234	6.8×10^{-7}	4.01×10^{-7}[g]

a. Abbreviations: µg/L = micrograms per liter; µCi/mL = microcuries per milliliter.
b. Total recoverable.
c. EPA National Secondary Drinking Water Standard (WSRC 1995a).
d. EPA Final Primary Drinking Standard (WSRC 1995a).
e. SCDHEC Final Primary Drinking Water Standard (WSRC 1995a).
f. Drinking Water Standards do not apply. Criterion 10 times a recently published 90th percentile detection limit was used (WSRC 1995a).
g. EPA Proposed Primary Drinking Water Standards (WSRC 1995a).

TC

Table 3.2-7. L-Area maximum reported groundwater parameters in excess of regulatory and SRS limits.[a]

Analyte	Concentration (µg/L for metals and organics; µCi/mL for radioisotopes)	Regulatory limit (µg/L for metals and organics; µCi/mL for radioisotopes)
Aluminum[b]	320	50[c]
Boron[b]	1,590	300[d]
Iron[b]	14,100	300[d]
Lead[b]	58	50[e]
Manganese[b]	771	50[c]
Tetrachloroethylene	17	5[d]
Total Organic Carbon	3.5×10^{-6}	10,000[f]
Nitrate-nitrite as Nitrogen	268,000	10,000[d]
Thallium[b]	7.4	2.0[d]
Tritium	5.4×10^{-4}	2.0×10^{-5}[d]
Non-volatile Beta	1.7×10^{-6}	5.0×10^{-8}[g]

a. Abbreviations: µg/L = micrograms per liter; µCi/mL = microcuries per milliliter.
b. Total recoverable.
c. EPA National Secondary Drinking Water Standard (WSRC 1995a).
d. EPA Final Primary Drinking Water Standard (WSRC 1995a).
e. SCDHEC Final Primary Drinking Water Standard (WSRC 1995a).
f. Drinking Water Standards do not apply. Criterion 10 times a recently published 90th percentile detection limit was used (WSRC 1995a).
g. EPA Interim Final Primary Drinking Water Standards (WSRC 1995a).

Table 3.2-8. P-Area maximum reported groundwater parameters in excess of regulatory and SRS limits.[a]

Analyte	Concentration (µg/L for metals and organics)	Regulatory limit (µg/L for metals and organics)
Aluminum[b]	19,900	50[c]
Iron[b]	22,200	300[d]
Manganese[b]	419	50[c]
Carbon tetrachloride	11	5[d]
Trichloroethylene	24	50[d]
Tetrachloroethylene	8.4	5[d]
Total organic halogens	79	50[e]
Tritium	7.7×10^{-2} Ci/mL	2.0×10^{-5}[d] µCi/mL
Strontium-90	1.7×10^{-6} Ci/mL	8.0×10^{-9}[d] µCi/mL

a. Abbreviations: µg/L = micrograms per liter; µCi/mL = microcuries per milliliter.
b. Total recoverable.
c. EPA National Secondary Drinking Water Standard (WSRC 1995a).
d. EPA Final Primary Drinking Water Standard (WSRC 1995a).
e. Drinking Water Standards do not apply. Criterion 10 times a recently published 90th percentile detection limit was used (WSRC 1995a).

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2×10^{-4} per year or about once every 5,000 years (Weber et al. 1998). Since operations began in 1953, nine confirmed tornadoes have occurred on or near the Site. Nothing more than light damage occurred, 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 crossed South Carolina, which resulted in a frequency of about one every 8 years (WSRC 1993). Because the SRS is about 100 miles (160 kilometers) inland, the winds associated with hurricanes have usually diminished below hurricane force [i.e., equal to or greater than a sustained wind speed of 75 miles per hour (33.5 meters per second)] before reaching the Site. Winds exceeding hurricane force have been observed only once at the SRS (Hurricane Gracie in 1959) (Shedrow 1993).

3.3.3 RADIOLOGICAL AIR QUALITY

DOE provides detailed summaries of radiological releases to the atmosphere from SRS operations, along with resulting concentrations and doses, in a series of annual environmental data reports. This section references several of those documents, which contain additional information. The information enables comparisons of current data with potential releases, concentrations, and doses associated with each alternative.

In the SRS region, airborne radionuclides originate from natural sources (terrestrial and cosmic), worldwide fallout, and Site operations. DOE maintains a network of air monitoring stations on and around the Site to determine concentrations of radioactive particulates and aerosols in the air (Arnett and Mamatey 1998b).

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Table 3.3-1 lists average and maximum atmospheric radionuclide concentrations at the SRS boundary and at background monitoring locations [100-mile (160-kilometer) radius] during 1997. Tritium is the only radionuclide from the SRS detected routinely in offsite air samples above background (control) concentrations (Cummins, Martin, and Todd 1990, 1991; Arnett et al. 1992; Arnett, Karapatakis, and Mamatey 1993, 1994; Arnett and Mamatey 1996; Arnett and Mamatey 1997; Arnett and Mamatey 1998b). Table 3.3-2 lists 1997 radionuclide releases from each major operational group of SRS facilities. All radiological impacts are within regulatory requirements.

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Figure 3.2-4. Maximum reported groundwater contamination at Savannah River Site.

Table 3.3-1. Radioactivity in air at SRS boundary and at 100-mile (160-kilometer) radius during 1997 (picocuries per cubic meter).[a]

Location	Tritium	Gross alpha	Gross beta	Cobalt-60	Cesium-137	Strontium-89,90	Plutonium-238	Plutonium-239
Site boundary								
Average[b]	11	9.8×10^{-4}	0.015	5.7×10^{-4}	1.5×10^{-4}	8.0×10^{-5}	(c)	(c)
Maximum[d]	65	0.0033	0.032	0.024	0.0073	3.6×10^{-4}	4.1×10^{-6}	7.0×10^{-6}
Background (100-mile radius)								
Average	3.2	0.0011	0.011	(c)	(c)	8.9×10^{-4}	6.9×10^{-6}	(c)
Maximum	5.4	0.0030	0.018	0.0073	0.0055	0.0019	4.2×10^{-5}	2.6×10^{-5}

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a. Source: Arnett and Mamatey (1998a).
b. The average value is the average value of the arithmetic means reported for the site perimeter sampling locations.
c. Below background levels.
d. The maximum value is the highest value of the maximums reported for the site perimeter sampling locations.

3.3.4 NONRADIOLOGICAL AIR QUALITY

The SRS is in the Augusta (Georgia) - Aiken (South Carolina) Interstate Air Quality Control Region. This region, which is designated a Class II area, is in compliance with National Ambient Air Quality Standards for criteria pollutants. Class II is the initial designation of any area that is not pristine; pristine areas include national parks or national wilderness areas. Criteria pollutants include sulfur dioxide, nitrogen oxides (reported as nitrogen dioxide), particulate matter (less than or equal to 10 microns in diameter), carbon monoxide, ozone, and lead (40 CFR 50).

DOE used the comprehensive emissions inventory data for 1996, which is the most recent available, to establish the baseline year for showing compliance with national and state air quality standards by calculating actual emission rates for existing sources of criteria pollutants. DOE based these emission rates on process knowledge, source testing, material balance, and EPA's Industrial Source Complex Air Dispersion Model.

SCDHEC has air quality regulatory authority over SRS. SCDHEC determines ambient air quality compliance based on air pollutant emissions and estimates of concentrations at the Site boundary based on atmospheric dispersion modeling. The SRS is in compliance with National Ambient Air Quality Standards for criteria pollutants and gaseous fluoride and with total suspended particulate standards, as required by SCDHEC Regulation R.61-62.5, Standard 2, "Ambient Air Quality Standards." Table 3.3-3 lists these standards and the results of the atmospheric dispersion modeling for baseline year 1996.

The SRS is in compliance with SCDHEC Regulation R.61-62.5, Standard 8, "Toxic Air Pollutants," which regulates the emission of 257 toxic air pollutants (WSRC 1994). DOE has identified emission sources for 139 of the 257 regulated air toxics; the modeled results indicate that the Site is in compliance with SCDHEC air quality standards. Table 3.3-4 lists toxic air pollutants that are the same as those the alternative actions described in this EIS would emit, and compares maximum downwind concentrations at the Site boundary for baseline year 1990, which is the most recent data available, to SCDHEC standards for toxic air pollutants.

Table 3.3-2. Radiological atmospheric releases by operational group for 1997.[a]

Radionuclide[b]	Half-life	Reactors	Separations[c]	Reactor materials	Heavy water	SRTC[d]	Diffuse and fugitive[e]	Total	
		Curies released							
Gases and vapors									
H-3 (oxide)	12.3 years	5.2×10^{3}	3.3×10^{4}		350		150	3.9×10^{4}	
H-3 (elem)	12.3 years		1.9×10^{4}					1.9×10^{4}	
H-3 Total	12.3 years	5.2×10^{3}	5.2×10^{4}		350		150	5.8×10^{4}	
C-14	5.73×10^{3} years		3.1×10^{-2}				1.9×10^{-8}	3.1×10^{-2}	
Kr-85	10.73 years		9.6×10^{3}					9.6×10^{3}	
I-129	1.57×10^{7} years		7.1×10^{-3}				1.2×10^{-7}	7.1×10^{-3}	
I-131	8.040 days		2.9×10^{-5}			2.98×10^{-5}		5.9×10^{-5}	
I-133	20.8 hours					4.92×10^{-4}		4.9×10^{-4}	
Particulates									
Na-22	2.605 years						1.1×10^{-9}	1.1×10^{-9}	
Mn-54	312.2 days						4.8×10^{-12}	4.8×10^{-12}	
Co-57	271.8 days		2.2×10^{-7}				1.0×10^{-9}	2.1×10^{-7}	
Co-58	70.88 days						1.7×10^{-12}	1.7×10^{-12}	TC
Co-60	5.271 years		3.5×10^{-7}				9.1×10^{-7}	1.3×10^{-6}	
Ni-59	7.6×10^{4} years						3.2×10^{-10}	3.2×10^{-10}	
Ni-63	100 years						2.3×10^{-9}	2.3×10^{-9}	
Zn-65	243.8 days						3.7×10^{-12}	3.7×10^{-12}	
Se-79	6.5×10^{4} years						2.2×10^{-10}	2.2×10^{-10}	
Sr-89,90[f]	29.1 years	1.8×10^{-3}	2.2×10^{-4}	4.2×10^{-5}	1.8×10^{-4}		8.2×10^{-5}	2.3×10^{-3}	
Zr-95	64.02 days						2.1×10^{-5}	2.1×10^{-5}	
Nb-95	34.97 days						1.6×10^{-15}	1.6×10^{-15}	
Tc-99	2.13×10^{5} years						3.6×10^{-8}	3.6×10^{-8}	
Ru-106	1.020 years						0.070	0.070	
Sn-126	1×10^{5} years						3.4×10^{-15}	3.4×10^{-15}	
Sb-124	60.2 days						3.4×10^{-12}	3.4×10^{-12}	
Sb-125	2.758 years						5.9×10^{-7}	5.9×10^{-7}	
Cs-134	2.065 years		1.4×10^{-6}				1.2×10^{-9}	1.4×10^{-6}	
Cs-137	30.17 years	2.5×10^{-4}	4.2×10^{-4}		2.9×10^{-6}		4.2×10^{-3}	4.9×10^{-3}	
Ba-133	10.53 years						3.0×10^{-12}	3.0×10^{-12}	
Ce-144	284.6 days		4.2×10^{-6}				6.1×10^{-6}	1.0×10^{-5}	
Pm-144	360 days						1.3×10^{-12}	1.3×10^{-12}	

Table 3.3-2. (Continued).

Radionuclide[b]	Half-life	Reactors	Separations[c]	Reactor materials	Heavy water	SRTC[d]	Diffuse and fugitive[e]	Total	
		Curies released							
Particulates (continued)									
Pm-147	2.6234 years						1.0×10^{-8}	1.0×10^{-8}	
Eu-152	13.48 years						5.3×10^{-9}	5.3×10^{-9}	
Eu-154	8.59 years		1.5×10^{-7}				6.4×10^{-6}	6.6×10^{-6}	
Eu-155	4.71 years		4.9×10^{-6}				1.7×10^{-6}	6.6×10^{-6}	
Ra-226	1.6×10^{3} years						1.2×10^{-8}	1.2×10^{-8}	
Ra-228	5.76 years						1.8×10^{-10}	1.8×10^{-10}	
Th-228	1.913 years						2.2×10^{-10}	2.2×10^{-10}	
Th-230	7.54×10^{4} years						2.0×10^{-10}	2.0×10^{-10}	
Th-232	1.40×10^{10} years						1.4×10^{-10}	1.4×10^{-10}	
Th-234	24.10 days						2.3×10^{-10}	2.3×10^{-10}	
Pa-231	3.28×10^{4} years						1.0×10^{-9}	1.0×10^{-9}	
Pa-234	6.69 hours						2.3×10^{-10}	2.3×10^{-10}	
U-233	1.592×10^{5} years						2.1×10^{-8}	2.1×10^{-8}	
U-234	2.46×10^{5} years		8.0×10^{-6}	4.0×10^{-6}			1.5×10^{-5}	2.7×10^{-5}	TC
U-235	7.04×10^{8} years		6.3×10^{-7}	6.4×10^{-7}			4.8×10^{-7}	1.8×10^{-6}	
U-236	2.342×10^{7} years						4.8×10^{-7}	4.8×10^{-7}	
U-238	4.47×10^{9} years		1.9×10^{-5}	1.7×10^{-6}			3.5×10^{-5}	5.6×10^{-5}	
Np-237	2.14×10^{6} years						1.4×10^{-9}	1.4×10^{-9}	
Np-239	2.35 days						2.2×10^{-7}	2.2×10^{-7}	
Pu-238	87.7 years		3.3×10^{-5}	4.4×10^{-9}			3.6×10^{-4}	3.9×10^{-4}	
Pu-239[g]	2.410×10^{4} years	2.9×10^{-4}	5.1×10^{-5}	6.9×10^{-6}	2.3×10^{-5}	2.5×10^{-6}	6.9×10^{-6}	3.8×10^{-4}	
Pu-240	6.56×10^{3} years						1.1×10^{-6}	1.1×10^{-6}	
Pu-241	14.4 years						5.2×10^{-5}	5.2×10^{-5}	
Pu-242	3.75×10^{5} years						3.7×10^{-11}	3.7×10^{-11}	
Am-241	432.7 years		1.4×10^{-5}	1.2×10^{-8}			8.7×10^{-7}	1.5×10^{-5}	
Am-243	7.37×10^{3}yr						1.8×10^{-5}	1.8×10^{-5}	
Cm-242	162.8 days						8.2×10^{-12}	8.2×10^{-12}	

Table 3.3-2. (Continued).

Radionuclide[b]	Half-life	Reactors	Separations[c]	Reactor materials	Heavy water	SRTC[d]	Diffuse and fugitive[e]	Total
		Curies released						
Particulates (continued)								
Cm-244	18.1 years		2.5×10^{-5}	2.0×10^{-10}			1.3×10^{-4}	1.5×10^{-4}
Cm-245	8.5×10^{3} years						1.9×10^{-12}	1.9×10^{-12}

a. Source: Arnett and Mamatey (1998a).
b. H = hydrogen (H-3 = tritium), C = carbon, Kr = krypton, I = iodine, Na = sodium, Mn = manganese, Co = cobalt, Ni = nickel, Zn = zinc, Se = selenium, Sr = strontium, Zr = zirconium, Nb = niobium, Tc = technetium, Ru = ruthenium, Sn = tin, Sb = antimony, Cs = cesium, Ba = barium, Ce = cerium, Pm = promethium, Eu = europium, Ra = radium, Th = thorium, Pa = protactinium, U = uranium, Np = neptunium, Pu = plutonium, Am = americium, Cm = curium.
c. Includes F- and H-Area releases.
d. SRTC = Savannah River Technology Center.
e. Estimated releases from minor unmonitored diffuse and fugitive sources.
f. Includes unidentified beta emissions.
g. Includes unidentified alpha emissions.

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Table 3.3-3. SRS baseline air quality for maximum potential emissions and observed ambient concentrations.

Pollutant	Averaging time	SCDHEC ambient standard ($\mu g/m^3$)[a]	Estimated SRS baseline concentration ($\mu g/m^3$)[b]
Criteria pollutants			
Sulfur dioxide (as SO_x)[c]	3-hr	1,300	1,200
	24-hr	365	350
	Annual	80	34
Total suspended particulates	Annual	75	67
Particulate matter (≤10 μm)[d]	24-hr	150	133
	Annual	50	25
Carbon monoxide	1-hr	40,000	10,000
	8-hr	10,000	6,900
Nitrogen dioxides (as NO_x)[e]	Annual	100	26
Lead	Calendar Quarterly mean	1.5	0.03
Ozone (as total VOCs)[f]	1-hr	235	NA[g]
Toxic/hazardous air pollutants			
Benzene	24-hr	150	3.9
Beryllium	24-hr	0.01	0.009
Biphenyl	24-hr	6	0.02
Mercury	24-hr	0.25	0.03
Methyl alcohol (methanol)	24-hr	1,310	0.9

TC

SO_x = oxides of sulfur; NO_x = oxides of nitrogen: VOCs = volatile organic compounds; NA = not available.

a. Source: SCDHEC Standard 2, "Ambient Air Quality Standards," and Standard 8, "Toxic Air Pollutants" (SCDHEC 1976).
b. Source: Hunter (1999). Concentration is the sum of modeled air concentrations using the permitted maximum potential emissions from the 1998 air emissions inventory for all SRS sources not exempted by Clean Air Act Title V requirements and observed concentrations from nearby ambient air monitoring stations.
c. Based on emissions for all oxides of sulfur (SO_x).
d. New NAAQS for particulate matter ≤2.5 microns (24-hour limit of 65 $\mu g/m^3$ and an annual average limit of 15 $\mu g/m^3$) will become enforceable during the life of this project.
e. Based on emissions for all oxides of nitrogen (NO_x).
f. New NAAQS for ozone (8 hours limit of 0.08 parts per million) will become enforceable during the life of this project.
g. Ambient concentrations of VOCs, which are precursors to ozone, can be used to provide a highly conservative bounding estimate for ozone but should not be used for explicit assessments of compliance with the ozone standard. Not all the VOCs emitted will result in the formation of ozone, and there is no method to directly correlate the two quantities. For purposes of estimating ozone concentrations from all SRS operations, no value for total VOCs is provided since the estimate would be overly conservative.

Table 3.3-4. Estimated 24-hour average ambient concentrations at SRS boundary - toxic air pollutants regulated by South Carolina from SRS sources.[a]

Pollutant[b]	Concentration ($\mu g/m^3$)[c]	Regulatory standard ($\mu g/m^3$)	Concentration as a percent of standard (%)
Benzene	31	150	20.70
Hexane	0.07	200	0.04
Nitric acid	6.70	125	5.40
Sodium hydroxide	0.01	20	0.05
Toluene	1.60	2,000	0.08
Xylene	3.80	4,350	0.09

$\mu g/m^3$ = micrograms per cubic meter.

a. Source: WSRC (1994).

b. Pollutants listed include air toxics of interest in relation to spent nuclear fuel management alternatives. (Section 5.2 addresses the effects of all air toxics.)

c. Based on actual emissions from existing SRS sources plus maximum potential emissions for sources permitted for construction through December 1992.

DOE measures nonradiological air emissions from SRS facilities at their points of discharge by direct measurement, sample extraction and measurement, or calculation of the emissions using process knowledge. Using monitoring data and meteorological information, DOE estimates the concentration of certain pollutants at the Site boundary. The Site is in compliance with National Ambient Air Quality Standards.

TC The Environmental Protection Agency approved revisions to the national ambient air quality standards for ozone and particulate matter that became effective on September 16, 1997. However, on May 14, 1999, in response to challenges filed by industry and others, the U.S. Court of Appeals for the District of Columbia Circuit issued a split opinion (2 to 1) directing

TC EPA to develop a new particulate matter standard (meanwhile reverting back to the previous PM_{10} standard) and ruling that the new ozone standard "cannot be enforced" (EPA 1999). The EPA has asked the U.S. Department of Justice to appeal this decision and take all judicial steps necessary to overturn the decision. Therefore, it is uncertain at this time when new ozone and particulate matter standards will become enforceable.

3.4 Ecological Resources

The U.S. Government acquired the land that became SRS in 1951. At that time, the Site was approximately two-thirds forested and one-third cropland and pastures. An extensive forest management program conducted by the Savannah River Natural Resources Management and Research Institute (SRI), which is part of the U.S. Forest Service, has converted many croplands and pastures to pine plantations. At present, more than 90 percent of the SRS is forested.

The Site provides more than 181,000 acres (734 square kilometers) of contiguous forested cover broken only by unpaved secondary roads, transmission line corridors in various stages of succession, a few paved primary roads, and scattered industrial facilities. Carolina bays, the Savannah River Swamp, and several relatively intact longleaf pine-wiregrass communities contribute to the biodiversity of the SRS and the entire region.

Under some of the alternatives described in Chapter 2, DOE proposes to construct and operate a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility at SRS to receive, characterize, condition, treat, package, and dry-store spent nuclear fuel before shipping it to

a geologic repository. If not located in an existing reactor building, the site for either of these facilities would cover approximately 15 acres (0.061 square kilometer), including the building footprint(s), construction area needs, and security requirements (WSRC 1996a).

As described in Chapter 2, this Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would be in L Area (preferred site), C Area, or P Area. Facilities to implement the New Processing Technology Alternative also could be located inside a reactor building, such as Building 105-L.

The proposed site for any new facility in L Area is a ridge that runs southwest-to-northeast approximately 0.5 mile (0.8 kilometer) from the Steel Creek floodplain. The site, which is wholly within the developed portion of L Area, is bounded by L Reactor to the west, a rail spur (L Line) to the north, and paved access roads to the east and south. The area consists of buildings, paved areas, graveled areas, and mowed turf grasses. The site is inside 6-foot (1.8-meter) security fences and has negligible value as wildlife habitat.

An upland pine stand is immediately east of the proposed site, adjacent to the fenced area. The stand is primarily slash pines (*Pinus elliotti*) that the Forest Service planted in the mid-1950s, with small areas of long-leaf (*P. palustris*) and loblolly pine (*P. taeda*) planted in the 1940s (SRFS 1997). Understory species include black cherry (*Prunus serotina*), wax myrtle (*Myrica cerifera*) and yellow jessamine (*Gelsemium sempervirens*). SRI manages forested areas such as this for timber production and wildlife.

Wildlife characteristically found in SRS pine plantations include toads (i.e., the southern toad, [*Bufo terrestris*]), lizards (e.g., the eastern fence lizard, [*Sceloporus undulatus*]), snakes (e.g., the black racer, [*Coluber constrictor*]), songbirds (e.g., the brown-headed nuthatch [*Sitta pusilla*], and the pine warbler [*Dendroica pinus*]), birds of prey (e.g., the sharp-shinned hawk [*Accipiter striatus*]), and a number of mammal species (e.g., the cotton mouse [*Peromyscus gossypinus*]), the gray squirrel [*Sciurus carolinensis*], the opossum [*Didelphis virginiana*], and the white-tailed deer [*Odocoileus virginianus*]) (Sprunt and Chamberlain 1970; Cothran et al. 1991; Gibbons and Semlitsch 1991; Halverson et al. 1997).

The proposed site for a new facility in C Area is on a plateau that rises between the floodplains of Fourmile Branch to the north and Castor Creek to the south. The entire site is inside the developed portion of C Area, surrounded by security fencing. The area consists of buildings, paved areas, graveled areas, and mowed turf grasses. A paved access road, a railroad spur, and two transmission lines cross the site. It provides little or no wildlife habitat. The areas immediately north and south of the site are forested, primarily with long-leaf and loblolly pine planted in the 1950s. The shrub layer contains young oaks (*Quercus* spp.) black cherry, hawthorne (*Crataegus* sp.), wax myrtle, and bear-grass (*Yucca filamentosa*). The wildlife species listed for L Area occur in these woods as well.

The proposed facility site in P Area is a broad hilltop above the headwaters of Steel Creek (to the west), Meyers Branch (to the south), and Lower Three Runs/Par Pond (to the east). The western two-thirds of the area (adjacent to the P-Area fence) is meadow-like, comprised mostly of lawn grasses and a few common forbs, such as low hop clover (*Trifolium dubium*) and smooth vetch (*Vicia dasycarpa*). The remainder of the area is wooded, with trees that appear to have regenerated since P Area was developed in the early 1950s. The canopy layer is dominated by laurel oak (*Quercus laurifolia*), water oak (*Q. nigra*), blackjack oak (*Q. marilandica*), mockernut hickory (*Carya alba*), and long-leaf pine. In the sub-canopy and shrub layer, species such as *Q. laevis* (turkey oak), huckleberry (*Vaccinium stamineum*), and hawthorne are well represented. Wooded areas to the north and east of the site are predominantly slash pines that were planted in the 1950s and loblolly pines that were planted in the 1980s (SRFS 1997). Because it is regularly mowed, the grassy area provides lim-

ited wildlife habitat. The wooded areas presumably provide habitat for many of the wildlife species mentioned above.

Under the Endangered Species Act of 1973 the Federal government provides protection to six species that occur on the SRS: American alligator (*Alligator mississippiensis*; threatened due to similarity of appearance to the endangered American crocodile), short-nosed sturgeon (*Acipenser brevirostrum*; endangered), bald eagle (*Haliaeetus leucocephalus*; threatened), wood stork (*Mycteria americana*; endangered), red-cockaded woodpecker (*Picoides borealis*; endangered), and smooth purple coneflower (*Echinacea laevigata*; endangered) (SRFS 1994). None of these species is known to occur on or near the proposed facility sites in L, C, P, F, or H Areas, which are located on previously disturbed areas (SRFS 1996).

3.5 Socioeconomics

Approximately 90 percent of the 1995 SRS workforce lived in the SRS region of influence which includes Aiken, Allendale, Bamberg, and Barnwell Counties in South Carolina, and Columbia and Richmond Counties in Georgia. *Socioeconomic Characteristics of Selected Counties and Communities Adjacent to the Savannah River Site* (HNUS 1997) contains additional information on the economic and demographic characteristics of the six-county region.

3.5.1 EMPLOYMENT

Between 1980 and 1990, total employment in the six-county region increased from 181,072 to 241,409, an average annual growth rate of approximately 2.9 percent. The unemployment rates for 1980 and 1990 were 7.3 percent and 4.7 percent, respectively (HNUS 1997). In 1994, regional employment was 243,854, an increase of only 1 percent since 1990. Over the next 10-year period, employment in the region is projected to increase at an average rate of slightly less than 1 percent per year, reaching approximately 264,000 by 2004 (HNUS 1997).

The increase in employment in the 1980s was spurred in part by the buildup in employment at

the SRS during the middle and late years of the decade, and in part by the improved national economy. The flat increases in regional employment since 1990 are the result of the mild national recession from 1990 to 1992, followed by the decreases in SRS employment, discussed below.

At the beginning of fiscal year 1996, employment at SRS was 16,625, approximately 7 percent of regional employment, with an associated annual payroll of approximately $634 million. This represents a decrease of 6,726 in SRS employment since 1992 and an associated payroll reduction of $466 million from more than $1.1 billion. Site employment declined through attrition by approximately 950 jobs between the fall of 1995 and the fall of 1996 and by another approximately 850 jobs in early 1997 through involuntary separations. By March 1998, the SRS workforce was reported at 14,014 persons (DOE 1998).

3.5.2 POPULATION

Based on state and Federal agency surveys and trends, the estimated 1998 population in the region of influence was 466,222. About 90 percent lived in Aiken (29 percent), Columbia (20 percent), and Richmond (41 percent) counties. The population in the region grew at an annual growth rate of about 6.5 percent between 1990 and 1998 (U.S. Bureau of the Census 1999). Columbia County, and to a lesser extent Aiken County, contributed to most of the growth due to in-migration from other region of influence counties and other states. Over the same period Bamberg and Barnwell counties experienced net outmigration. In 2000, the population in the six-county region is expected to be approximately 498,900. Over the next 10-year period, the regional population should grow at a projected rate of 1 to 2 percent per year, reaching approximately 533,400 by 2010. 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 31.8 years (HNUS 1997).

3.5.3 COMMUNITY CHARACTERISTICS

Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations* (February 11, 1994), directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse human health or environmental effects of their programs, policies, and activities on minority and low-income populations. Executive Order 12898 also directs the Administrator of EPA to convene an interagency Federal Working Group on Environmental Justice.

The Working Group has provided guidance to Federal agencies on criteria for identifying disproportionately high and adverse human health or environmental effects on minority and low-income populations (EPA 1998). In addition, the Council on Environmental Quality, in consultation with EPA and other Federal agencies, has developed guidance for identifying and addressing environmental justice concerns during the National Environmental Policy Act (NEPA) process (CEQ 1998). DOE has based the environmental justice analysis in this document on those guidance documents. Further, in coordination with the Working Group, DOE is developing internal guidance for implementing the Executive Order.

Potential offsite health impacts from the proposed action would result from releases to the air and to the Savannah River downstream of the SRS. For air releases, DOE performed standard population dose analyses on a 50-mile (80-kilometer) radius because reasonably foreseeable dose levels beyond that distance would be negligible. For liquid releases, the region of interest includes areas that draw drinking water from the river (Beaufort and Jasper Counties in South Carolina and Effingham and Chatham Counties in Georgia).

The analysis included data (U.S. Bureau of the Census 1990a,b) for populations in census tracts with at least 20 percent of their area in the 50-mile radius and all tracts from Beaufort and

Jasper Counties and Effingham and Chatham Counties, which are downstream of the Site. DOE used data from each census tract in this combined region to identify the racial composition of communities and the number of persons characterized by the U.S. Bureau of the Census as living in poverty. The combined region contains 247 census tracts, 99 in South Carolina and 148 in Georgia.

Tables 3.5-1 and 3.5-2 list racial and poverty characteristics, respectively, of the population in the combined region. Table 3.5-1 indicates a total population of more than 993,000 in the area. Of that population, approximately 618,000 (62.2 percent) are white. In the minority population, approximately 94 percent are African American; the remainder are small percentages of Asian, Hispanic, and Native American persons. Figure 3.5-1 shows the distribution of minorities by census tract areas in the SRS region.

Executive Order 12898 does not define minority populations. One approach to a definition is to identify communities that contain a simple majority of minorities (greater than or equal to 50 percent of the total community population). A second approach, proposed by EPA for environmental justice purposes, defines minority communities as those that have higher-than-average (over the region of interest) percentages of minority persons (EPA 1994). The shading patterns in Figure 3.5-1 indicate census tracts where (1) minorities constitute 50 percent or more of the total population, or (2) minorities constitute between 35 percent and 50 percent of the total population. For this analysis, DOE has adopted the second, more expansive, approach to identify minority communities.

The combined region has 80 tracts (32.4 percent) where minority populations constitute 50 percent or more of the total population. In an additional 50 tracts (13.5 percent), minorities constitute between 35 and 50 percent of the population. These tracts are distributed throughout the region, although there are more toward the south and in the immediate vicinities of Augusta and Savannah, Georgia.

Table 3.5-1. General racial characteristics of population in SRS region of interest.[a]

State	Total population	White	African American	Hispanic	Asian	Native American	Other	Minority	Percent minority[b]
South Carolina	418,685	267,639	144,147	3,899	1,734	911	355	151,046	36.08
Georgia	574,982	350,233	208,017	7,245	7,463	1,546	478	224,749	39.09
Total	993,667	617,872	352,164	11,144	9,197	2,457	833	375,795	37.82

a. Source: U.S. Bureau of the Census (1990a).
b. People of color population divided by total population.

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Table 3.5-2. General poverty characteristics of population in SRS region of interest.[a]

Area	Total population	Persons living in poverty[b]	Percent living in poverty
South Carolina	418,685	72,345	17.28
Georgia	574,982	96,672	16.81
Total	993,667	169,017	17.01

a. Source: U.S. Bureau of the Census (1990b).
b. Families with income less than the statistical poverty threshold, which in 1990 was 1989 income of $8,076 for a family of two.

Low-income communities are those in which 25 percent or more of the population is characterized as living in poverty (EPA 1993). The U.S. Bureau of the Census defines persons in

poverty as those whose income is less than a "statistical poverty threshold." This threshold is a weighted average based on family size and the age of the persons in the family. The baseline threshold for the 1990 census was a 1989 income of $8,076 for a family of two.

Table 3.5-2 indicates that in the SRS region, more than 169,000 persons (17 percent of the population) are characterized as living in poverty. In Figure 3.5-2, shaded census tracts identify low-income communities. In the region, 72 tracts (29.1 percent) are low-income communities. These tracts are distributed throughout the region of analysis, but primarily to the south and west of SRS.

3.6 Cultural Resources

Through a cooperative agreement, DOE and the South Carolina Institute of Archaeology and Anthropology, University of South Carolina, conduct the Savannah River Archaeological Research Program to provide on the SRS services required by Federal law for the protection and management of archaeological resources. Ongoing research programs work in conjunction with the South Carolina State Historic Preservation Officer. They provide theoretical, methodological, and empirical bases for assessing site significance using the compliance process specified by law. Archaeological investigations usually begin through the Site Use Program, which requires a permit for clearing land on the SRS.

The archaeological research has provided considerable information about the distribution and content of archaeological and historic sites on SRS. Savannah River archaeologists have examined SRS land since 1974. To date they have examined 60 percent of the 300-square-mile (800-square kilometer) area and recorded more than 1,200 archaeological sites (HNUS 1997). Most (approximately 75 percent) of these sites are prehistoric.

The activities associated with the proposed action and alternatives for spent nuclear fuel management at SRS that could affect cultural

Figure 3.5-1. Distribution of minorities by census tract in SRS region of analysis.

Figure 3.5-2. Low-income census tracts in the SRS region of analysis.

resources are the use of one of the three sites for the proposed Transfer and Storage Facility or Transfer, Storage, and Treatment Facility.

The sites are in reactor areas (L, C, and P) within 100 to 400 yards (91 to 366 meters) of the reactor buildings. The Savannah River Archaeological Research Program has not examined any areas in and immediately around the reactors. Construction of these facilities took place before the enactment of Federal regulations to protect historic resources. Archaeological resources in the footprints of the three preferred sites would be unlikely to have survived reactor construction, although 1951 aerial photographs show that the C- and L-Area sites had homeplaces before the development of the SRS in the early 1950s (Sassaman 1997a,b).

The potential for prehistoric sites in the preferred locations is limited. The P-Area site is in archaeology site density Zone 2, which has moderate potential for prehistoric archaeological sites of significance. The L-Area site is in archaeological site density Zone 3, which has the least potential for prehistoric sites of significance. C Area is divided between Zones 2 and 3. However, in all cases, reactor construction activities probably destroyed or severely damaged any prehistoric deposits (Sassaman 1997a,b).

3.7 Public and Worker Health

3.7.1 PUBLIC RADIOLOGICAL HEALTH

Because there are many sources of radiation in the human environment, evaluations of radioactive releases from nuclear facilities must consider all ionizing radiation to which people are routinely exposed.

Doses of radiation are expressed as millirem (mrem), rem (1,000 millirem), and person-rem (which is the average individual doses times the population).

An individual's radiation exposure in the vicinity of SRS amounts to approximately 357 millirem per year, which is comprised of natural background radiation from cosmic, terrestrial, and internal body sources, radiation from medical diagnostic and therapeutic practices, weapons test fallout, consumer and industrial products, and nuclear facilities. Figure 3.7-1 shows the relative contributions of each source to people living near SRS. All radiation doses mentioned in this EIS are effective dose equivalents; internal exposures are committed effective dose equivalents.

Releases of radioactivity to the environment from SRS account for less than 0.1 percent of the total annual average environmental radiation dose to individuals within 50 miles (80 kilo-meters) of the Site. Natural background radiation contributes about 293 millirem per year, or 82 percent of the annual dose of 357 millirem received by an average member of the population within 50 miles of the Site. Based on national averages, medical exposure accounts for an additional 14.8 percent of the annual dose, and combined doses from weapons test fallout, consumer and industrial products, and air travel account for about 3 percent (NCRP 1987a).

Other nuclear facilities within 50 miles (80 kilometers) of SRS include a low-level waste disposal site operated by Chem-Nuclear Systems, Inc., near the eastern Site boundary and Georgia Power Company's Vogtle Electric Generating Plant, directly across the Savannah River from the Site. In addition, Carolina Metals, Inc., which is northwest of Boiling Springs in Barnwell County, processes depleted uranium.

South Carolina Nuclear Facility Monitoring - Annual Report 1992 (SCDHEC 1992) documents that the Chem-Nuclear and Carolina Metals facilities do not influence radioactivity levels in the air, precipitation, groundwater, soil, or vegetation. Plant Vogtle began commercial operation in 1987: 1992 releases produced an annual dose of 0.17 millirem to the maximally exposed individual at the plant boundary and a total population dose within a 50-mile (80-kilometer) radius of 0.057 person-rem (NRC 1996).

Figure 3.7-1. Major sources of radiation exposure in the vicinity of the Savannah River Site.

In 1997, releases of radioactive material to the environment from SRS operations resulted in a maximum individual dose of 0.05 millirem per year in the west-southwest sector of the Site boundary from atmospheric releases, and a maximum dose from liquid releases of 0.13 millirem per year, for a maximum total annual dose at the boundary of 0.18 millirem (Arnett and Mamatey 1998b). The maximum dose to downstream consumers of Savannah River water – 0.07 millirem per year – occurred to users of the Port Wentworth and the Beaufort-Jasper public water supplies (Arnent amd Mamatey 1998b).

In 1990 the population within 50 miles (80 kilometers) of the Site was approximately 620,100. The collective effective dose equivalent to that population in 1997 was 2.2 person-rem from atmospheric releases. The 1990 population of 65,000 people using water from the Cherokee Hill Water Treatment Plant near Port Wentworth, Georgia, and the Beaufort-Jasper Water Treatment Plant near Beaufort, South Carolina, received a collective dose equivalent of 2.4 person-rem in 1997 (Arnett and Mamatey 1998b). Population statistics indicate that cancer caused 23.2 percent of the deaths in the United States in 1994 (CDC 1998). If this percentage of deaths from cancer continues, 23.2 percent of the U.S. population will contract a fatal cancer from all causes. Thus, in the population of 620,100 within 50 miles of SRS, 143,863 persons will be likely to contract fatal cancers from all causes. The total population dose from SRS of 4.6 person-rem (2.2 person-rem from atmospheric pathways plus 2.4 person-rem from water pathways) could result in 0.0023 additional latent cancer death in the same population [based on 0.0005 cancer death per person-rem (NCRP 1993)].

3.7.2 PUBLIC NONRADIOLOGICAL HEALTH

The hazards associated with the alternatives described in this EIS include exposure to nonradiological chemicals in the form of water and air pollution (see Sections 3.2 and 3.3). Table 3.3-3 lists ambient air quality standards and concentrations for selected pollutants. The purpose of these standards is to protect the public health and welfare. The concentrations of pollutants from SRS sources, listed in Table 3.3-2, are lower than the standards. Section 3.2 discusses water quality in the SRS vicinity.

3.7.3 WORKER RADIOLOGICAL HEALTH

One of the major goals of the SRS Health Protection Program is to keep worker exposures to radiation and radioactive material as low as reasonably achievable (ALARA). Such a program must evaluate both external and internal exposures with the goal to minimize the total effective dose equivalent. An effective ALARA program must also balance minimizing individual worker doses with minimizing the collective dose of workers in a group. For example, using many workers to perform small portions of a task would reduce the individual worker dose to low levels. However, frequent worker changes would make the work inefficient, resulting in a significantly higher collective dose to all the workers than if fewer had received slightly higher individual doses.

SRS worker doses have typically been well below DOE worker exposure limits. DOE set administrative exposure guidelines at a fraction of the exposure limits to help enforce doses that are as low as reasonably achievable. For example, the current DOE worker exposure limit is 5,000 millirem per year, and the 1997 SRS ALARA administrative control level for the whole body is 500 millirem per year. Every year DOE evaluates the SRS ALARA administrative control levels and adjusts them as needed.

Table 3.7-1 lists maximum and average individual doses and SRS collective doses from 1989 to 1998.

Table 3.7-1. SRS annual individual and collective radiation doses.[a]

Year	Number with measurable dose	Average individual worker dose (rem)[b]	Site worker collective dose (person-rem)
1989	12,363	0.070	863
1990	11,659	0.065	753
1991	8,391	0.055	459
1992	6,510	0.054	352
1993	5,202	0.051	264
1994	6,284	0.050	315
1995	4,846	0.053	256
1996	4,736	0.053	252
1997	3,327	0.050	165
1998	3,163	0.052	166

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a. Adapted from: DOE (1996b); WSRC (1997, 1998, 1999a).
b. The average dose includes only workers who received a measurable dose during the year.

3.7.4 WORKER NONRADIOLOGICAL HEALTH

Industrial hygiene and occupational health programs at the SRS deal with all aspects of worker health and relationship with the work environment. The objective of an effective occupational health program is to protect employees from hazards in their work environment. To evaluate these hazards, DOE uses routine monitoring to determine employee exposure levels to hazardous chemicals.

Exposure limit values are the basis of most occupational health codes and standards. If an overexposure to a harmful agent does not exist, that agent generally does not create a health problem.

The Occupational Safety and Health Administration (OSHA) has established Permissible Exposure Limits to regulate worker exposure to hazardous chemicals. These limits refer to airborne concentrations of substances and represent conditions under which nearly all workers could receive repeated exposures day after day without adverse health effects.

Table 3.7-2 lists the estimated maximum and average annual concentrations of existing OSHA-regulated workplace pollutants modeled in and around existing SRS facilities. Estimated concentration levels for existing OSHA-regulated workplace pollutants are less than the OSHA Permissible Exposure Limits for all contaminants, with the exception of nitrogen dioxide (as nitrogen oxide) and nitric acid. The large nitrogen dioxide exceedance (a 15-minute average of 406 $mg/m3$ compared to the OSHA Permissible Exposure Limit of 9 mg/m^{-3}) is based on modeling assumptions with maximum potential emissions for diesel units including back-up units operating at ground-level for limited periods (Stewart 1997). The nitric acid value also is based on maximum potential emissions related to conventional processing activities. Actual emissions are expected to be below regulatory limits.

DOE has established industrial hygiene and occupational health programs for the processes covered by this EIS and across the SRS to protect the health of workers from nonradiological hazards.

3.8 Waste and Materials

3.8.1 WASTE MANAGEMENT

This section describes the waste generation baseline that DOE uses in Chapter 4 to gauge the relative impact of each SNF management alternative on the overall production of waste at SRS and on DOE's capability to manage such waste.

Table 3.7-2. Estimated maximum annual concentrations (milligrams per cubic meter) of workplace pollutants regulated by Occupational Safety and Health Administration.[a]

Pollutant	OSHA PEL[b] (mg/m^3)	Time period	Concentrations (mg/m^3) Maximum 8-hour average	Concentrations (mg/m^3) Annual average
Carbon monoxide	55	8 hours	10	0.53
Nitrogen dioxide (as NO_x)	9	Ceiling limit[c]	406[d]	2.3
Total particulates	15	8 hours	0.95	0.06
Sulfur dioxide (as SO_x)	13	8 hours	0.63	0.05
Hexane	1,800	8 hours	1.5	0.08
Nitric acid	5	8 hours	11	0.34
Sodium hydroxide	2	8 hours	<0.01	<0.01
Xylene	435	8 hours	136	14.5

a. Source: Stewart (1997).
b. OSHA Permissible Exposure Limits (PEL).
c. Ceiling limits are permissible exposure limits that a facility cannot exceed at any time.
d. 15-minute average.

SRS generates six basic classes of waste – low-level radioactive, high-level radioactive, hazardous, mixed (low-level radioactive and hazardous), transuranic and alpha, and sanitary (nonhazardous, nonradioactive) – which this EIS considers because they are possible byproducts of SNF management. The following sections describe the waste classes. Table 3.8-1 lists projected total waste generation volumes for fiscal years 1999 through 2029 (a 30-year time period encompassing most of the time period of the scenarios addressed in this EIS). TC

Tables 3.8-2 through 3.8-4 provide an overview of the existing and planned facilities that DOE expects to use in the storage, treatment, and disposal of the various waste classes.

3.8.1.1 Low-Level Radioactive Waste

EC DOE Order 435.1 (Radioactive Waste Management) defines low-level radioactive waste as radioactive waste that cannot be classified as high-level waste, spent nuclear fuel, transuranic waste, or byproduct material, and that does not have any constituents that are regulated under the Resource Conservation and Recovery Act (RCRA).

At present, DOE uses a number of methods for treating and disposing of low-level waste at SRS, depending on the waste form and activity. Approximately 41 percent of this waste is low in activity and can be treated at the Consolidated Incineration Facility. In addition, DOE could volume-reduce these wastes by compaction, supercompaction, smelting, or repackaging (DOE 1995c). After volume reduction, DOE would package the remaining low-activity waste and place it in either shallow land disposal or vault disposal in E Area.

DOE places low-level wastes of intermediate activity and some tritiated low-level wastes in E Area intermediate activity vaults, and will store long-lived low-level waste (e.g., spent deionizer resins) in the long-lived waste storage buildings in E Area, where they will remain until DOE determines their final disposition.

3.8.1.2 Low-Level Mixed Waste

DOE Order 435.1 defines low-level mixed waste as low-level radioactive waste that contains material listed as hazardous under RCRA or that exhibits one or more of the following hazardous waste characteristics: ignitability, corrosivity, TC

reactivity, or toxicity. It includes such materials

Table 3.8-1. Total waste generation forecast for SRS (cubic meters).[a,b]

	Waste Class				
Inclusive Dates	Low-level	High-level	Hazardous	Mixed low-level	Transuranic and alpha
1999 to 2029	180,299	14,129	6,315	3,720	6,012

Source: Derived from Halverson (1999).

as tritiated mercury, tritiated oil contaminated with mercury, other mercury-contaminated compounds, radioactively contaminated lead shielding, equipment from the tritium facilities in H Area, and filter paper takeup rolls from the M-Area Liquid Effluent Treatment Facility.

As described in the *Approved Site Treatment Plan* (DOE 1996c), storage facilities for low-level mixed waste are in several different SRS areas. These facilities are dedicated to solid, containerized, or bulk liquid waste and all are approved for this storage under RCRA as interim status or permitted facilities or as Clean Water Act-permitted tank systems. Several treatment processes described in the *Approved Site Treatment Plan* (DOE 1996c) exist or are planned for low-level mixed waste. These facilities, which are listed in Table 3.8-3, include the Consolidated Incineration Facility, the M-Area Vendor Treatment Process, and the Hazardous Waste/Mixed Waste Containment Building.

Depending on the nature of the waste remaining after treatment, DOE plans to use either shallow land disposal or RCRA-permitted hazardous waste/mixed waste vaults for disposal.

3.8.1.3 High-Level Waste

High-level radioactive waste is highly radioactive material from the processing of SNF that contains a combination of transuranic waste and fission products in concentrations that require permanent isolation. It includes both liquid waste produced by processing and solid waste derived from that liquid (DOE 1988).

At present, DOE stores high-level waste in carbon steel and reinforced concrete underground tanks in the F- and H-Area tank farms. The high-level waste undergoes volume reduction by evaporation, and the resulting high activity precipitate is incorporated in borosilicate glass at the Defense Waste Processing Facility Vitrification Facility. The remaining low-activity salt solution is treated and disposed of at the Saltstone Manufacturing and Disposal Facility. Both processes are described in the *Final Supplemental Environmental Impact Statement, Defense Waste Processing Facility* (DOE 1994).

DOE has committed to complete closure by 2022 of the 24 HLW tank systems that do not meet the secondary containment requirements in the Federal Facility Agreement (WSRC 1999a). During waste removal, DOE will retrieve as much of the stored HLW as can be removed using the existing waste transfer equipment. The sludge portion of the retrieved waste will be treated in treatment facilities and vitrified at DWPF. The salt portion of the retrieved waste (processed and treated) will be treated at one of the salt disposition facilities being evaluated in the High-Level Waste Salt Disposition Alternatives EIS (DOE 1999b) and either vitrified at DWPF or disposed as grout in Z Area.

3.8.1.4 Sanitary Waste

Sanitary waste is solid waste that is neither hazardous, as defined by RCRA, nor radioactive. It consists of salvageable material and material that is suitable for disposition in a municipal sanitary landfill. Sanitary waste streams include

Table 3.8-2. Planned and existing waste storage facilities.[a]

Storage facility	Location	Capacity	Original waste stream[b]						Status
			Low-level	High-level	Transuranic	Alpha[c]	Hazardous	Mixed Low-level	
Long-lived waste storage buildings	E Area	140 m^3/ bldg	X						One exists.
Containerized mixed waste storage	Buildings 645-2N, 643-29E, 643-43E, 316-M, and Pad 315-4M	4754 m^3						X	DOE plans to construct additional storage buildings, similar to 643-43E, as necessary.
Liquid mixed waste storage	DWPF Organic Waste Tank (S Area) SRTC Mixed Waste Tanks Liquid Waste Solvent Tanks (H Area) Burial Ground Solvent Tanks (E Area)	9531 m^3						X	The Burial Ground Solvent Tanks are currently undergoing closure. The H-Area Liquid Waste Solvent Tanks were constructed as a replacement.
High-level waste tank farms	F and H Area	(d)		X					50 underground tanks are currently used for storage[e].
Failed equipment storage vaults	Defense Waste Processing Facility (S Area)	300 m^3		X					Two exist; DOE plans approximately 12 additional vaults.
Glass waste storage buildings	Defense Waste Processing Facility (S Area)	2,286 canisters		X					One exists; a second is planned for construction in 2007.
Hazardous waste storage facility	Building 710-B Building 645-N Building 645-4N Waste Pad 1 (between 645-2N and 645-4N) Waste Pad 2 (between 645-4N and 645-N) Waste Pad 3 (east of 645-N)	2,501 m^3					X		Currently in use. No additional facilities are planned, as existing space is expected to adequately support the short-term storage of hazardous wastes awaiting treatment and disposal.
Building 316-M	Building 316-M	117 m^3					X		Currently in use. No additional facilities are planned.
Transuranic waste storage pads	E Area	(f)			X	X		X	19 pads exist; 10 additional pads may be constructed by 2006.

DWPF = Defense Waste Processing Facility.
SRTC = Savannah River Technology Center.

a. Sources: DOE (1994, 1995a, 1995b, 1996a).
b. Sanitary waste is not stored at SRS, thus it is not addressed in this table.
c. Currently, alpha waste is handled and stored as transuranic waste.
d. Currently the High-level Waste Tanks contain approximately 130,600 m^3 of high-level waste. This is almost 90 percent of the usable capacity.
e. Twenty-three of these tanks do not meet secondary containment requirements and have been scheduled for waste removal.
f. Transuranic Pad storage capacities depends on the packaging of the waste and the configuration of packages on the pads.

Table 3.8-3. Planned and existing waste treatment processes and facilities.[a]

Waste Treatment Facility	Waste Treatment Process	Waste type							Status
		Low-level	High-level	Transuranic	Alpha[b]	Hazardous	Mixed low-level	Sanitary	
Consolidated Incineration Facility	Incineration	X				X	X		Began treating waste summer 1997
Offsite facility	Smelting	X							Currently ongoing
Defense Waste Processing Facility	Vitrification		X						Currently operational
Defense Waste Processing Facility	Stabilization						X		Currently operational
Replacement high-level waste evaporator[c]	Volume Reduction		X						Radioactive operation anticipated in March 2000
M-Area Vendor Treatment Facility	Vitrification						X		Undergoing Closure
Treatment at point of waste stream origin	Macroencapsulation						X		As feasible based on waste and location
Non-Alpha Vitrification Facility	Vitrification	X				X	X		Plan to begin operations in 2006
INEEL[d] Waste Engineering Development Facility	Amalgamation/ Stabilization						X		Developing shipping/ treatment schedules
Offsite facility	Offsite Treatment and Disposal					X			Currently ongoing
Offsite facility	Decontamination						X		Plan to begin shipment in FY2000
Various onsite and offsite facilities[e]	Recycle/Reuse	X				X	X	X	Currently ongoing
Alpha Vitrification Facility	Vitrification				X				Under evaluation as a potential process
Existing DOE facilities	Repackaging/ Treatment			X					Transuranic waste strategies are still being finalized
M-Area Air Stripper	Air Stripping					X			Currently operational
F- and H-Area Effluent Treatment Facility	Effluent Treatment	X							Currently operational

a. Sources: DOE (1994, 1995a, 1995b, 1996a); WSRC (1995a, 1995b, 1996b); and Odum (1995).
b. Currently, alpha waste is handled as transuranic waste. After it is assayed and separated, most will be treated and disposed of as low-level or mixed low-level waste.
c. Evaporation precedes treatment at the Defense Waste Processing Facility and is used to maximize high-level waste storage capacity.
d. Idaho National Engineering and Environmental Laboratory.
e. Various waste streams have components (e.g., silver, lead, freon, paper) that might be recycled or reused. Some recycling activities might occur onsite, while other waste streams are directed offsite for recycling. Some of the recycled products are released for public sale, while others are reused onsite.

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Table 3.8-4. Planned and existing waste disposal facilities.[a]

Disposal facility	Location	Capacity (m^3)	Original waste stream[b]						Status
			Low-level	High-level	Transuranic	Hazardous	Mixed Low-level	Sanitary	
Shallow land disposal trenches	E Area	(c)	X						Four have been filled; up to 58 more may be constructed.
Low-activity vaults	E Area	30,500/vault	X						One vault exists and one additional is planned.
Intermediate-activity vaults	E Area	5,300/vault	X						Two vaults exist and five more may be constructed.
Hazardous waste/mixed waste vaults	NE of F Area	2,300/vault				X	X		RCRA permit application submitted for 10 vaults. At least 11 additional vaults may be needed.
Saltstone Disposal Facility	Z Area	80,000/vault[d]	X						Two vaults exist and approximately 13 more are planned.
Three Rivers Landfill	SRS Intersection of SC 125 and Rd. 2	NA						X	Current destination for SRS sanitary waste.
Burma Road Cellulosic and Construction Waste Landfill	SRS Intersection of C Rd. and Burma Rd	NA						X	Current destination for demolition/construction debris. DOE expects to reach permit capacity in 2008.
Waste Isolation Pilot Plant (WIPP)	New Mexico	175,600			X				EPA certification of WIPP completed in April 1998.
									RCRA certification finalized in 1999.[e]
Federal repository	See Status	NA		X					Proposed Yucca Mountain, Nevada site is currently under investigation.

RCRA = Resource Conservation and Recovery Act.
NA = Not Available.

a. Sources: DOE (1994, 1995a, 1995b, 1995c, 1996a, 1996c); WSRC (1995a and 1996b).
b. After alpha waste is assayed and separated from the transuranic waste, DOE plans to dispose of it as low-level or mixed low-level waste so it is not addressed separately here.
c. Various types of trenches exist including engineered low-level trenches, greater confinement disposal boreholes and engineered trenches, and slit trenches. The different trenches are designed for different waste types, are constructed differently, and have different capacities.
d. This is the approximate capacity of a double vault. One single vault and one double vault have been constructed. Future vaults are currently planned as double vaults.
e. SRS is scheduled for WIPP certification audit in 2000, after which WIPP could begin receiving SRS waste.

such items as paper, glass, discarded office material, and construction debris (DOE 1994).

Sanitary waste volumes have declined due to recycling and the decreasing SRS workforce. DOE sends sanitary waste that is not recycled or reused to the Three Rivers Landfill on SRS. The SRS also continues to operate the Burma Road Cellulosic and Construction Waste Landfill to dispose of demolition and construction debris.

3.8.1.5 Hazardous Waste

Hazardous waste is nonradioactive waste that SCDHEC regulates under RCRA and corresponding state regulations. Waste is hazardous if the EPA lists it is as such or if it exhibits the characteristic(s) of ignitability, corrosivity, reactivity, or toxicity. SRS hazardous waste streams consist of a variety of materials, including mercury, chromate, lead, paint solvents, and various laboratory chemicals.

At present, DOE stores hazardous wastes in three buildings and on three solid waste storage pads that have RCRA permits. Hazardous waste is sent to offsite treatment and disposal facilities, and could be treated at the Consolidated Incineration Facility in the future. DOE also plans to continue to recycle, reuse, or recover certain hazardous wastes, including metals, excess chemicals, solvents, and chlorofluorocarbons. Wastes remaining after treatment might be suitable for either shallow land disposal or disposal in the Hazardous/Mixed Waste Disposal Vaults (DOE 1995c).

3.8.1.6 Transuranic and Alpha Waste

Transuranic waste contains alpha-emitting transuranic radionuclides (those with atomic weights greater than 92) that have half-lives greater than 20 years at activities exceeding 100 nanocuries per gram (DOE 1988). At present, DOE manages low-level alpha-emitting waste with activities between 10 and 100 nanocuries per gram, referred to as alpha waste, as transuranic waste at SRS.

The SRS Waste Management EIS (DOE 1995c) describes the handling and storage of transuranic and alpha waste at the SRS. This consists primarily of providing continued safe storage until treatment and disposal facilities are available.

The *Strategic Plan for Savannah River Site Transuranic Waste* (WSRC 1996b) defines the future handling, treatment and disposal of the SRS transuranic and alpha waste stream. Eventually, DOE plans to ship the transuranic and mixed transuranic waste to the Waste Isolation Pilot Plant in New Mexico for disposal.

Before disposition, DOE plans to assay the wastes stored on the pads and segregate the alpha waste. Vitrification is an option for at least part of the mixed alpha waste (DOE 1996b). Following assay, DOE could dispose of much of the alpha waste as either mixed low-level or low-level waste.

3.8.2 HAZARDOUS MATERIALS

The *Savannah River Site Tier Two Emergency and Hazardous Chemical Inventory Report* for 1998 (WSRC 1999b) lists more than 79 hazardous chemicals that were present at SRS at some time during the year in amounts that exceeded the minimum reporting thresholds [10,000 pounds (4,536 kilograms) for hazardous chemicals and 500 pounds (227 kilograms) or less for extremely hazardous substances]. Four of the 79 are extremely hazardous substances under the Emergency Planning and Community Right-to-Know Act of 1986. The actual number and quantity of hazardous chemicals present on the Site and at individual facilities changes daily as a function of use and demand.

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CHAPTER 4. ENVIRONMENTAL IMPACTS

This chapter describes potential environmental impacts from construction, operation, and accidents associated with the proposed action and its alternatives. Section 4.1 describes the operational impacts of each alternative within the scope of this environmental impact statement (EIS). Section 4.2 describes risks to members of the public and onsite workers from potential facility accidents associated with the management of spent nuclear fuel (SNF) at the Savannah River Site (SRS). Section 4.3 describes impacts that could result from construction activities associated with SNF management at SRS. The purpose of the information presented in this chapter is to provide comparisons among alternatives. For new facilities, this information is based on DOE's best estimates of these facilities' operational characteristics. These data are not intended to be used for safety analysis purposes or compared to safety documents such as a Safety Analysis Report.

As discussed in Section 2.3.2, the Department of Energy (DOE) has identified three candidate sites for the potential construction of a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility: (1) the east side of L Area inside the facility fence, (2) the southeast side of C Area inside the facility fence, and (3) the northeast side of P Area. In addition, the facility could be constructed on a site inside the F-Area or H-Area fence or in an existing reactor building such as Building 105-L.

In most instances, implementing the technology options described in Chapter 2 would result in the same or very similar environmental impacts, regardless of location. If, during the preparation of this EIS, analyses indicated that a technology option would produce different environmental impacts at one of the candidate sites, DOE analyzed the site that would have the greatest impact (the bounding site). The analysis of the atmospheric releases of radioactivity described in the air resources and public and worker health sections is based on the assumption that emissions from a Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would occur in C Area. Releases from C Area would result in higher estimated radiation doses to members of the public than releases from L or P Area (i.e., C Area would result in doses to the maximally exposed offsite individual approximately 1.7 times higher than those in L Area and 1.1 times higher than those in P Area). All other impacts would be independent of location.

The impacts reported in this chapter are based on the entire SNF inventory described in Chapter 1 and Appendix C. However, as noted in Section 1.3, some foreign reactor operators may not participate in DOE's program of accepting U.S.-origin SNF. This reduction in receipts could potentially impact the amounts of fuel in Groups B, D, and E. Therefore, the amounts of fuel to be managed in those fuel groups could be less than the amounts assumed for the calculations in Chapter 4. DOE believes that annual impacts for normal operations, construction impacts, and accident impacts would be unaffected by modest reductions in the expected fuel inventory. The annual impacts are based on the maximum year's impacts; decreasing the foreign fuel shipments may lessen the number of years of fuel handling, conditioning, or treatment, but would not affect the maximum annual impact. SNF accidents usually involve small amounts of fuel and thus are insensitive to the total inventory. Construction impacts are similarly insensitive to the reduction in total fuel inventory that could occur. Eleven environmental impact measures are based on activities that occur over the entire period of analysis. These impacts would be sensitive to reductions in fuel receipts. Where applicable, the tables in this chapter explain how to adjust reported impacts for potentially reduced fuel receipts.

4.1 Impacts from Normal Operations

This section describes environmental impacts that could result from operational activities associated with SNF management at SRS for existing and new facilities. Because the only potential impacts to geologic and cultural resources would occur during construction (see Section 4.3), Section 4.1 does not consider geologic or cultural resource impacts. DOE does not anticipate a significant increase in employment due to the implementation of any technology options (Table 4.1-1). The existing site work force should be sufficient to provide the necessary operations and support personnel; therefore, there would be no socioeconomic impacts from operations under any technology.

Table 4.1-1. Estimated operational staffing for any of the technology options.

Technology option	Operations personnel	Support personnel	Total personnel
Melt and Dilute	200	200	400
Mechanical Dilution	175	175	350
Repackage and Prepare to Ship	75	75	150
Vitrification	317	317	634
Electrometallurgical	238	238	476
Conventional Processing	300	300	600
Continued Wet Storage	80	80	160

Source: Bickford et al. 1997.

DOE used the following process to estimate the impacts associated with new facilities/processes. First, DOE identified the facilities that would be needed to implement each of the technologies described in Chapter 2 (see Table 2-4). Next, DOE identified the major systems required within each facility for each technology. DOE then identified the energy sources, potential waste and effluent streams, and sources of potential radiation exposure associated with each of these major systems. These results were then compared to similar processes with which DOE has operational experience to determine the relative magnitude of the impact. These impacts were

presented as annual impacts; integrated impacts were then calculated as described below in Section 4.1.1.

DOE does not expect normal operations to have any appreciable impacts on ecological resources. Impacts would be limited to minor disturbances of animals in undeveloped areas adjacent to SNF management facilities caused by increased movement and noise from personnel, vehicles, and equipment. However, these impacts would be negligible under all proposed technology options because they would occur in areas where industrial activities already exist. Impacts to potential human receptors from normal releases of radioactive and nonradioactive contaminants to the environment would be small for any of the technologies under consideration (Section 4.1.1.3). Therefore, these releases would not be likely to produce measurable effects on nearby plant and animal communities or to accumulate in aquatic or terrestrial ecosystems.

4.1.1 IMPACTS OF TECHNOLOGY OPTIONS

This section describes the environmental impacts of each technology. The analysis covers the environmental impacts of actions over the 38-year period from 1998 through 2035 and presents both maximum annual impacts from these technologies and estimated total impacts over the entire period. For example, the discussions of water and air resources present maximum annual radiation doses to members of the public from liquid and airborne emissions associated with each technology and compares the resulting values to Federal limits. The section on public and worker health, on the other hand, presents radiation doses to members of the public from liquid and airborne emissions over the entire implementation period. The waste generation and utilities and energy sections also present impacts over the entire period of analysis (1998-2035). EC

To estimate total impacts, DOE identified the activities necessary to implement each technology, the amount of time required for each step (*phase*) of the technology option, and the annual impacts likely to occur during each phase. DOE summed the annual impacts over the entire duration of the phase, together with other phases needed to implement that option. For the Conventional Processing option, DOE used historic data for F- and H-Canyon operations to estimate the time needed to process the entire inventory of each type of fuel (McWhorter 1997). For the other technology options with a treatment phase, DOE used engineering judgments to estimate the duration of this phase for each fuel group. Appendix E describes the assumed durations for each phase. If annual impact data (i.e., utilities and energy, waste generation, and worker radiation dose) for each type of fuel were not available, DOE assumed that the fraction of the impact attributable to each type of fuel would be equal to the fraction of that fuel's fissile mass to the total fissile mass of SNF in the scope of this EIS. DOE derived the annual impact calculations from the available data (Bickford et al. 1997) based on the total radionuclide inventory for each type of fuel. Appendix C contains the radionuclide inventories, using a "reference fuel assembly" i.e., a conservative estimate of the radionuclide and curie content for an SNF assembly designed to bound the characteristics of fuel assigned to SRS. The engineering report that provides data upon which the impacts presented in this chapter are based (Bickford et al. 1997) is available for review at the DOE public reading room in Aiken, South Carolina.

4.1.1.1 Water Resources

This section describes the effects of normal operations associated with the technologies to SRS waters. All process water would come from groundwater. None of the technologies require much water to process the fuels. At most, less than 6,000 liters per year (equivalent to 1,585 gallons per year) would be required. The SRS annually withdraws more than 5×10^9 liters of groundwater (DOE 1997).

As discussed below, the only technology that would result in discharges of radionuclides or nonradioactive hazardous materials to surface water would be conventional processing. The

major sources of liquid effluents from facilities associated with conventional processing would be process cooling water and steam condensate that could contain small quantities of radionuclides and chemicals. Conventional processing would use wastewater treatment facilities and other equipment designed for full production (i.e., five production reactors, two separation facilities, and other industrial facilities) loads. Therefore, capacities would be sufficient to handle the liquid effluents and other secondary waste associated with conventional processing.

Liquid effluents associated with the SNF technologies would use existing wastewater treatment facilities and outfalls described in Section 3.2.1.3. Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility (CSWTF) and discharged through an existing NPDES outfall (G-10). Because technology options would not increase the number of permanent SRS employees, the CSWTF treatment rates would not be affected, and it would continue to meet the requirements of the SRS NPDES permit.

DOE evaluated in the Programmatic SNF EIS (DOE 1995b) the potential impacts to groundwater from a direct leak to the subsurface from a breach in a storage pool during routine operations. Because basin water could contain some radionuclides but would not contain any toxic or harmful chemicals, the following evaluation addresses only the consequences of radionuclide releases. The analysis conservatively assumed a 5-gallon (19-liter) per-day leak as a result of secondary containment or piping failure at the Receiving Basin for Offsite Fuels, L-Reactor Disassembly Basin, or a new wet receipt basin in a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility. The analysis assumed further that the leak would go undetected for 1 month.

The reliability and sensitivity of the leak detection devices at a new wet receipt basin would be equal to or superior to those required by the U.S. Nuclear Regulatory Commission (NRC 1975) for SNF storage facilities in commercial nuclear power plants. Constant process monitoring, mass balance, and facility design (including double-walled containment of vessels and piping) also would be used by DOE to limit operational releases from a new wet receipt facility to near zero.

A leak from the Receiving Basin for Offsite Fuels, or the L-Reactor Disassembly Basin, could result in the introduction of radionuclide-contaminated water into the ground at depths as much as 44 feet (13.4 meters) below grade. Such a release would go directly to the uppermost aquifer (Upper Three Runs), which at SRS is not suitable for use as a drinking water source because of its low yield and the presence of contaminants. Any contaminants would move through the Upper Three Runs and Gordon aquifers and ultimately discharge to SRS streams. The processes governing the plume movement (i.e., the hydraulic conductivity, hydraulic gradient, and effective porosity of aquifers in F, H, and the Reactor 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) would mitigate impacts to surface- or groundwater resources. Localized contamination of groundwater in the surface aquifer could occur in the immediate vicinity of the storage facility. However, this aquifer is not used as a source of drinking water. DOE concludes that no radionuclide contamination of deeper confined aquifers that are sources of onsite or offsite drinking water would be likely to occur from a leak in a storage basin.

The aquifer used as the primary source for drinking water is separated from the shallower aquifers by a confining unit. The hydraulic pressure of the lower aquifer is greater than that of the overlying aquifer. Therefore, water flows from the lower to the upper aquifer. This upward flow would prevent the downward migration of released contaminants.

4.1.1.1.1 Radiological Impacts

With the exception of conventional processing which is the maximum impact alternative, none of the technologies proposed in this EIS is likely to result in measurable increases in radionuclides released to water (Bickford et al. 1997). No other proposed technology would have a process discharge to surface waters.

The prolonged storage of SNF in the basins (i.e., the No-Action Alternative) could lead to a higher rate of fuel failures and releases to basin water, but probably would not affect routine releases (i.e., those from national pollutant discharge elimination system [NPDES] permitted outfalls). DOE would maintain water quality by monitoring basin water, deionizing basin water using resin beds, and stabilizing leaking assemblies.

Calculations of radiological doses through water pathways based on these releases are supported by the use of LADTAPXL, a spreadsheet version of the LADTAP II computer code developed by the U.S. Nuclear Regulatory Commission (NRC) to estimate radiation doses associated with normal reactor system liquid effluent releases to individuals, populations, and biota (Hamby 1991). LADTAP II uses the models in NRC Regulatory Guide 1.109 (NRC 1977) to calculate doses received from water and fish ingestion and from recreational water activities. Parameters used to calculate dose for the maximally exposed individual are consistent with regularly published SRS environmental reports (e.g., Arnett and Mamatey 1996).

Any radionuclide releases to surface water resulting from the technologies would be to SRS streams that discharge to the Savannah River. For all technology options, the ingestion of fish contaminated with cesium-137 would contribute most of the exposure to both the maximally exposed individual and the population. Plutonium and uranium isotopes ingested with drinking water would be smaller contributors for the approximately 70,000 people served by water treatment plants near Port Wentworth, Georgia (60,000) and Beaufort, South Carolina (10,000) (Arnett and Mamatey 1996). Table 4.1-2 lists both the maximally exposed individual dose and the collective dose due to liquid releases to the 620,100-person population surrounding SRS. TC

Table 4.1-2. Estimated maximum incremental annual dose to hypothetical maximally exposed individual and 620,100-person population surrounding SRS due to liquid releases from Conventional Processing.

Fuel group	MEI dose (millirem)	Population dose (person-rem)
A. Uranium and Thorium Metal Fuels	4.2×10^{-5}	2.4×10^{-4}
B. Materials Test Reactor-Like Fuels	0.042	0.14
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0.014	0.047
D. Loose Uranium Oxide in Cans	1.4×10^{-3}	4.7×10^{-3}
E. Higher Actinide Targets	NA	NA
F. Non-Aluminum-Clad Fuels	NA	NA

NA = Technology is not applicable to this fuel type.
HEU = Highly Enriched Uranium.
LEU = Low Enriched Uranium.
MEI = Maximally Exposed Individual.

4.1.1.1.2 Nonradiological Impacts

This assessment compared chemical releases with applicable water quality standards. These standards are based on the preservation of aquatic biota populations, human health, and aesthetics (i.e., taste and odor). Figure 3.2-1 shows that conventional processing activities would not occur in the 100-year floodplain. DOE would treat sanitary waste generated by any of the alternatives in this EIS in existing sewage treatment fa-

cilities; discharges from these facilities would continue to meet NPDES permit limits.

Activities associated with the New Packaging Technology options and all new treatment options under the New Processing Technology, including Melt and Dilute, Mechanical Dilution, Vitrification, and Electrometallurgical Treatment, would conform to current regulatory standards, and would not have nonradiological waterborne releases (Bickford et al. 1997). Under conventional processing, process cooling water treatment would result in releases of the following concentrations from F Area to Upper Three Runs:

- Nitrate - 40 micrograms per liter
- Ammonia - 30 micrograms per liter
- Manganese - 10 micrograms per liter
- Uranium - 20 micrograms per liter
- Nickel - 50 micrograms per liter
- Chromium - 20 micrograms per liter
- Aluminum - 200 micrograms per liter
- Copper - 10 micrograms per liter
- Zinc - 70 micrograms per liter

Similar or lower concentrations would be released from H Area with the exception of those for nitrate and ammonia, which would be 100 and 500 micrograms per liter, respectively.

Although proposed or final Federal drinking water standards do not apply to discharges, the SRS discharge concentrations would not exceed these standards. The discharges would also comply with South Carolina Water Quality Standards contained in South Carolina Regulation R.61-68. In general, the release concentrations would be no greater than those currently measured in Upper Three Runs and Fourmile Branch (Arnett 1996), with the exception of zinc and ammonia; however, zinc concentrations in the discharge would be only a small fraction of the South Carolina Water Quality Standards, which are based on the taste and odor of drinking water. Ammonia concentrations in the discharge (only H-Area releases would increase current stream concentrations) would be well within state standards. Lead, nickel, and chromium generally were not detected in Upper Three Runs and Fourmile Branch in 1995.

4.1.1.2 Air Resources

This section describes incremental air quality impacts from nonradiological and radiological emissions for the operation of each technology option for each fuel group; this description includes impacts to on- and offsite individuals and populations.

This analysis presents results in terms of ground-level air concentrations for nonradiological constituents and radiation dose for radionuclides because these are the best measures of potential adverse human health effects.

4.1.1.2.1 Nonradiological Emissions

DOE estimated nonradiological emission rates for each technology option (Bickford et al. 1997) and used them with the meteorological data described in Section 3.3.1 to estimate site boundary and noninvolved worker concentrations. This analysis assumed average meteorological conditions.

Onsite Concentrations

The purpose of this analysis is to estimate air concentrations to which SRS workers not involved in SNF management and related operations would be exposed. Atmospheric emissions would occur from F or H Area (conventional processing), L-Reactor Disassembly Basin and the Receiving Basin for Offsite Fuels (continued wet storage), and the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility. To determine impacts to noninvolved workers, the analysis used a generic location 2,100 feet (640 meters) from the release in the direction of the plume of greatest concentration. The 2,100-foot criterion is based on NRC guidance. Also, the use of this distance ensures consistency between this and previous SRS EISs.

The analysis assumed that operational nonradiological releases would be from the same release stack as radiological releases. In addition, this EIS does not include onsite concentrations at distances greater than 2,100 feet; the analysis considered such concentrations and found that they would be less than those at 2,100 feet.

Tables F-1 through F-10 in Appendix F list estimated air concentrations above baseline (i.e., incremental increases) resulting from nonradiological atmospheric emissions associated with SNF fuel groups. No incremental atmospheric emissions above the baseline presented in Chapter 3 would be associated with Repackage and Prepare to Ship, the only option applicable to the non-aluminum-clad fuels. The air quality regulatory standards listed in Tables F-6 through F-10 in Appendix F are applicable to the Site boundary concentration from all SRS emissions. While these standards are included only for reference, all the incremental concentrations from SNF activities would be at least two orders of magnitude less than any of the corresponding standards except those for nitric acid, oxides of nitrogen, and gaseous fluorides emitted during conventional processing or vitrification of fuel Group B. The concentrations would range from less than 1 percent to about 55 percent of the offsite standard (for nitrogen oxides). If a new facility or a major modification to an existing facility were being considered, new permitting actions would be required as part of the Clean Air Act Title V permit compliance requirements. Under the current Title V permit, SRS would have to conduct a Prevention of Significant Deterioration review, since the nitrogen oxide levels exceed the 25 μm per cubic meter per year threshold of NO_2 for a Class II area. In addition, there would be a requirement for ambient monitoring to verify emission levels once the process began.

Offsite Concentrations

This analysis presents projected maximum offsite nonradiological incremental air concentrations in much the same way it presents the onsite concentrations. The estimated maximum incremental concentrations listed in Tables F-6 through F-10 in Appendix F would occur at the SRS boundary for emissions associated with SNF. The air quality regulatory standards listed in the tables are applicable to the Site boundary concentrations from all SRS emissions. All the incremental concentrations are at least three orders of magnitude less than any of the corresponding standards except those for oxides of nitrogen and gaseous fluorides emitted during conventional processing or vitrification. The concentrations ranged from less than 1 percent to about 2 percent of the offsite standard.

4.1.1.2.2 Radiological Emissions

DOE estimated airborne radionuclide emission rates for each technology option (Bickford et al. 1997), and used them with the meteorology data from Section 3.3.1 as inputs to the SRS com-

puter models MAXIGASP and POPGASP (Hamby 1994) to determine doses to onsite (noninvolved worker) and offsite (hypothetical maximally exposed individual) recipients and the surrounding population (620,000 persons) within a 50-mile (80-kilometer) radius of the center of the Site (Simpkins 1996). The analysis uses the meteorological data to determine annual average concentrations in air. The values presented in Tables 4.1-3, 4.1-4, and 4.1-5 represent current reactor-area emissions (including two SNF wet basins).

Onsite Doses

Atmospheric doses to the noninvolved worker represent the radiological exposures of a hypothetical worker who is nearby but not involved in SNF operations. Table 4.1-3 lists the estimated maximum incremental annual doses to noninvolved workers from atmospheric emissions of radionuclides for each viable technology option for each fuel group. The EPA limit of 10 millirem per year (40 CFR Part 61, Subpart H) is a point of comparison for these doses. (In fact, this limit is applicable to offsite individuals from sitewide airborne releases; see Chapter 5). The highest incremental dose to the noninvolved worker would be 0.27 millirem (from Melt and Dilute, Vitrification, or Electrometallurgical Treatment of Materials Test Reactor-like Fuels). Incremental doses to the noninvolved worker from all viable options would be 3 percent or less of the national emission standards for hazardous air pollutants (NESHAP) limit.

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There would be no pathways for exposure of personnel inside SNF management facilities from atmospheric releases of radioactivity. Section 4.1.1.3 discusses radiation doses to SNF management workers, including from in-facility airborne releases of radioactivity.

Offsite Doses

Atmospheric doses to the hypothetical maximally exposed offsite individual assume a person who resides at the SRS boundary at the point of maximum exposure. Every member of the public would have a dose less than that received by this individual. Table 4.1-4 lists the estimated maximum incremental annual dose to this individual from atmospheric emissions of radionuclides for each technology option for each fuel group. As with the doses to noninvolved workers, the NESHAP limit of 10 millirem per year (40 CFR Part 61, Subpart H) is a point of comparison. The maximum incremental annual dose from any technology option for a given fuel group would be 0.033 millirem per year (from Melt and Dilute, Vitrification, or Electrometallurgical Treatment of Materials Test Reactor-like Fuels), a factor of 300 less than the EPA limit.

Table 4.1-5 lists the estimated maximum incremental annual population dose (the collective dose to the entire population around SRS) for each viable option. The maximum incremental annual population dose from any option would be 1.2 person-rem per year (from Melt and Dilute, Vitrification, or Electrometallurgical Treatment of Materials Test Reactor-like Fuels).

4.1.1.3 Worker and Public Health

This section discusses potential radiological and nonradiological health effects to SRS workers and the surrounding public from the technology options for the management of SNF; it does not include impacts of potential accidents, which are discussed in Section 4.2. DOE based its calculations of health effects from the air- and waterborne radiological releases on (1) the dose to the hypothetical maximally exposed individual

in the public; (2) the collective dose to the population within a 50-mile (80-kilometer) radius around the SRS (approximately 620,000 people); (3) the collective dose to workers involved in implementing a given alternative (i.e., the workers involved in SNF management activities); and (4) the dose to the maximally exposed noninvolved worker (i.e., SRS employees who may work in the vicinity of the SNF management facilities but are not directly involved in SNF work). All radiation doses mentioned in this EIS are effective dose equivalents; internal exposures are committed effective dose equivalents. This section presents total impacts for the entire length of time necessary to implement each technology, using the durations listed in Appendix E. The annual impacts attributable to each phase were multiplied by the duration of that phase. The impacts from all phases were summed to calculate the total impact for the technology. This discussion characterizes health effects as additional lifetime latent cancer fatalities likely to occur in the general population around SRS and in the population of workers who would be associated with the options.

4.1.1.3.1 Radiological Health Effects

Radiation can cause a variety of health effects in people. The major effects that environmental and occupational radiation exposures could cause are delayed cancer fatalities, which are called latent cancer fatalities because the cancer can take many years to develop and cause death.

To relate a dose to its effect, DOE has adopted a dose-to-risk conversion factor of 0.0004 latent cancer fatality per person-rem for workers and 0.0005 latent cancer fatality per person-rem for the general population (NCRP 1993). The factor for the population is slightly higher due to the presence of infants and children who might be more sensitive to radiation than workers, who are, generally speaking, healthy adults.

DOE uses these conversion factors to estimate the effects of exposing a population to radiation. For example, in a population of 100,000 people exposed only to background radiation (0.3 rem per year), DOE would calculate 15 latent cancer fatalities per year caused by radiation (100,000 persons $\times$ 0.3 rem per year $\times$ 0.0005 latent cancer fatality per person-rem).

Calculations of the number of latent cancer fatalities associated with radiation exposure might not yield whole numbers and, especially in environmental applications, might yield values less than 1. For example, if a population of 100,000 were exposed only to a dose of 0.001 rem to each person, the collective dose would be 100 person-rem, and the corresponding number of latent cancer fatalities would be 0.05 (100,000 persons $\times$ 0.001 rem $\times$ 0.0005 latent cancer fatality per person-rem).

DOE also has employed these concepts in estimating the effects of radiation exposure to a single individual. For example, consider the effects of exposure to background radiation over a lifetime. The number of latent cancer fatalities corresponding to an individual's exposure over a (presumed) 72-year lifetime at 0.3 rem per year would be 0.011 latent cancer fatality (1 person $\times$ 0.3 rem per year $\times$ 72 years $\times$ 0.0005 latent cancer fatality per person-rem).

This number should be interpreted in a statistical sense; that is, the estimated effect of background radiation exposure to the exposed individual is a 1.1-percent lifetime chance that the individual might incur a latent fatal cancer. Vital statistics on mortality rates for 1994 (CDC 1996) indicate that the overall lifetime fatality rate in the United States from all forms of cancer is about 23.4 percent (23,400 fatal cancers per 100,000 deaths).

These factors, which DOE uses in this EIS to relate radiation exposure to latent cancer fatalities, are based on the *Recommendations of the International Commission on Radiation Protection* (ICRP 1991). They are consistent with the factors used by the U.S. Nuclear Regulatory Commission in its rulemaking *Standards for Protection Against Radiation* (10 CFR Part 20). The factors apply if the dose to an individual is less than 20 rem and the dose rate is less than 10

rem per hour. At doses greater than 20 rem, the factors used to relate radiation doses to latent cancer fatalities are doubled. At much higher dose rates, prompt effects, rather than latent cancer fatalities, would be the primary concern.

In addition to latent cancer fatalities, other health effects could result from environmental and occupational exposures to radiation; these include nonfatal cancers among the exposed population and genetic effects in subsequent generations. Previous studies have concluded that these effects are less probable than fatal cancers as consequences of radiation exposure (ICRP 1991). Dose-to-risk conversion factors for nonfatal cancers and hereditary genetic effects (0.0001 per person-rem and 0.00013 per person-rem, respectively) are substantially lower than those for fatal cancers. This EIS presents estimated effects of radiation only in terms of latent cancer fatalities because that is the major potential health effect from exposure to radiation. Estimates of nonfatal cancers and hereditary genetic effects can be estimated by multiplying the radiation doses by the effects dose-to-risk conversion factors.

DOE expects minimal worker and public health impacts from the radiological consequences of managing SNF under any of the technology options, as well as Continued Wet Storage. However, some options would result in increased radiological releases. Public radiation doses include doses from airborne releases (Section 4.1.1.2) and liquid releases (Section 4.1.1.1). Table 4.1-6 lists incremental radiation doses estimated for the public (maximally exposed individual and collective population dose) and corresponding incremental latent cancer fatalities, for each fuel group and technology option.

The values in Tables 4.1-6 and 4.1-8 for the No-Action Alternative represent current reactor-area emissions (including two SNF wet basins) for the entire period of analysis. The values for the other alternatives would be incremental above these baseline values. Summing these baseline and incremental values would be conservative, however, because there would not be two SNF wet basins operating over the entire 38-year period of analysis.

DOE based estimated worker doses on past operating experience and the projected durations for implementation of the alternative actions (Bickford et al. 1997). For the maximally exposed worker, DOE assumed that no worker would receive an annual dose greater than 500 millirem from any option because SRS uses the 500-millirem value as an administrative limit for normal operations; that is, an employee who receives an annual dose approaching the administrative limit normally is reassigned to duties in a nonradiation area. (Note: If DOE privatized the Transfer and Storage Facility or treatment operations, the licensee would adopt NRC worker dose limits, and administrative limits could be subject to adjustment.) Tables 4.1-7 and 4.1-8 estimate radiation doses for the collective population of workers who would be directly involved in implementing the options and for the noninvolved worker (a worker not directly involved with implementing the option but located 2,100 feet [640 meters] from the SNF facility) for each fuel group and technology option. These tables also list the latent cancer fatalities likely attributable to the doses. TC

Of the fuels considered for treatment (all except higher actinide targets and non-aluminum clad fuel), the highest expected radiological health effects to the public generally would occur under conventional processing. The single exception would be fewer latent cancer fatalities predicted for the population from the conventional processing of uranium and thorium metal fuels (Table 4.1-6). For the noninvolved workers, the conventional processing of Groups C and D fuels would result in the greatest radiological health effects. No measurable incremental increases would be likely for the higher actinide targets or the non-aluminum-clad fuels for any option because the only options applied to those groups are repackaging and continued wet storage. The estimated collective dose for workers who would be directly involved in managing SNF (Table 4.1-7) depends largely on the difference in the number of workers involved in each option and

not on the difference in the amount of radioactivity.

The estimated number of latent cancer fatalities in the public listed in Table 4.1-6 can be compared to the projected number of fatal cancers (145,100) in the public around the SRS from all causes (as discussed in Section 3.7.1). Similarly, the estimated number of latent cancer fatalities in the worker population can be compared to the number in the worker population from all causes (approximately 23.4 per- cent; see Section 3.7.1). In all cases, the incremental impacts from the options would be negligible.

4.1.1.3.2 Nonradiological Health Effects

DOE evaluated the range of chemicals to which the public and workers would be exposed due to SNF management activities and expects minimal health impacts from nonradiological exposures. Section 4.1.1.1.1 discusses offsite chemical concentrations from air emissions. DOE estimated worker impacts and compared them to Occupational Safety and Health Administration (OSHA) permissible exposure limits (PELs) or ceiling limits for protecting worker health, and concluded that all impacts would be well below the limits.

OSHA limits (29 CFR Part 1910.1000) are time-weighted average concentrations that a facility cannot exceed during a prescribed duration of a 40-hour week. The facility cannot exceed OSHA ceiling concentrations during any part of the workday. These exposure limits refer to airborne concentrations of substances and represent conditions under which nearly all workers could be exposed day after day without adverse health effects. However, because of the wide variation in individual susceptibility, a small percentage of workers could experience discomfort from some substances at concentrations at or below the permissible limit. Table 4.1-9 summarizes the values of Permissible Exposure Limits that DOE compared to the data in Tables F-1 through F-5 in Appendix F.

4.1.1.4 Waste Generation

This section presents waste generation estimates for each technology option and fuel group that DOE considers in this EIS. Tables 4.1-10 through 4.1-13 list these estimates. For each technology option, this analysis considered three handling phases as potential sources of waste: wet storage (pretreatment storage), treatment or conditioning, and dry storage (post-treatment storage pending final disposition). The period and waste generation rate associated with each phase varied depending on the fuel group and the technology. As discussed above, DOE summed waste volumes from each phase; the values listed in the tables represent the total projected waste volumes for each technology option in a given fuel group.

DOE used the annual waste generation rates to calculate the estimates in the tables (Bickford et al. 1997); the rates are based on applicable current and past SRS operations or on process

Table 4.1-9. Permissible Exposure Limits (milligrams per cubic meter) of nonradiological air pollutants regulated by the Occupational Safety and Health Administration.[a]

Pollutant	Averaging time	OSHA PEL[b]
Carbon monoxide	8 hours	55
Nitrogen oxides	1 hour	9[c]
Sulfur dioxide	8 hours	13
Carbon dioxide	8 hours	9,000
Nitric acid	8 hours	5

a. Source: 29 CFR Part 1910.1000.
b. Occupational Safety and Health Administration (OSHA) permissible exposure limit (PEL).

c. OSHA ceiling limit not to be exceeded at any time during the workday.

knowledge for new treatment technologies. The operating history that was the basis for these estimates would maximize projected waste generation rates. As described in Section 3.8, the Site generates several types of waste (high-level, transuranic, mixed, hazardous, low-level, and sanitary). Wastes generated by SNF management activities would be comparable to wastes the SRS currently handles and would, therefore, not require unique treatment, storage, or disposal actions. This section does not consider sanitary waste, the production of which would be in direct proportion to the number of employees, because none of the technologies would increase the number of permanent Site employees.

DOE has implemented an aggressive waste minimization and pollution prevention program at SRS at the sitewide level and for individual organizations and projects. As a result, significant reductions have been achieved in the amounts of wastes discharged into the environment and sent to landfills, resulting in significant cost savings.

To implement a waste minimization and pollution prevention program at the SNF management facilities, DOE would characterize waste streams and identify opportunities for reducing or eliminating them. Emphasis would be placed on minimizing the largest waste stream, low-level L1-5 waste, through source reduction and recycling. Selected waste minimization practices could include: (1) process design changes to reduce the potential for spills and to minimize contamination areas, (2) decontamination of equipment to facilitate reuse, (3) recycling metals and other usable materials, especially during the construction phase of the project, (4) preventive maintenance to extend process equipment life, (5) modular equipment designs to isolate potential failure elements to avoid changing out entire units, and (6) use of non-toxic or less toxic materials to prevent pollution and minimize hazardous and mixed waste streams.

EC The following sections describe the differences in waste generation by waste type among the SNF management technologies considered in this EIS.

4.1.1.4.1 High-Level Waste

SRS reports high-level waste as liquid high-level waste, and in the related quantities of equivalent Defense Waste Processing Facility (DWPF) canisters and saltstone. The volume estimates for liquid high-level waste reported in Table 4.1-10 are for volumes as they leave the process and enter the high-level waste tanks. While it is necessary to consider this volume when evaluating the interim storage of high-level waste in the tank farms, the volume of liquid high-level waste is not meaningful when considering the storage and disposition of final waste forms. The liquid waste is evaporated and concentrated in the high-level waste tanks. The generation of secondary waste in the high-level waste tanks and DWPF, including waste generated as a result of activities described in this SNF EIS, is evaluated in the DWPF Supplemental EIS (DOE 1994). Therefore, capacity for management of SNF secondary waste in the tank farms and DWPF is provided within the scope of DWPF operations. DWPF canisters and saltstone are the product of liquid high-level waste treatment and evaporation and would be the basis for final storage and disposition considerations. Because the production of saltstone and DWPF canisters from a given liquid waste volume are generally proportional, this discussion applies equally to DWPF canisters and saltstone. For Conventional Processing, DWPF canisters would be the only product to be disposed in a geologic repository. TC

Conventional Processing is the only option that would generate significant quantities of high-level waste during the treatment phase. Each option would produce high-level waste during the wet storage phase and technologies such as melt and dilute, that require off-gas collection systems, would also produce high-level waste, but the quantity produced generally would be much lower than that associated with Conventional Processing. The waste generated during wet storage and new technology processing operations would not meet the formal definition of high-level waste (waste resulting from the processing of SNF), but would consist of such items as deionizer backwash and off-gas collection

products, which the SRS typically manages (or would manage) as high-level waste. The lengthy period associated with continued wet storage generally would make it the second largest producer of high-level waste. For the higher actinide targets, Conventional Processing was not considered, making Continued Wet Storage the greatest potential for high-level waste production. The volumes of high-level waste generated by the other options would vary depending on the duration of storage and the amount of fissile material in the fuel, but would be fairly comparable within a given fuel type and substantially less than the volumes associated with conventional processing. In addition, the condition of the fuel would influence the high-level waste generation rate (i.e., fuel in poor condition would result in higher generation of deionizer backwash).

Based on the capacities of the high-level waste tank farms and the current volume of high-level waste in storage (see Table 3.8-2), these projected high-level waste volumes probably would not require additional treatment and storage facilities beyond those currently available at SRS. DOE bases this conclusion on continued removal and treatment of the existing tank farm inventory. DWPF would be available to treat these projected high-level waste volumes.

4.1.1.4.2 Transuranic Waste

For all applicable fuel types, conventional processing would produce the largest volume of transuranic waste due to a higher generation rate and a longer processing time. Conventional processing of all applicable fuel groups would generate 3660 cubic meters of transuranic waste which is 29 percent of the total SRS transuranic waste generation forecast (Table 3.8-1). The next largest quantity that could be generated would be from the Vitrification and Electrometallurgical Treatments of all applicable fuel groups. Those technologies would generate 700 cubic meters of transuranic waste over the life of the project, which is less than 6 percent of the total SRS transuranic waste generation forecast. These two technologies would produce 9 to 37 percent of that produced by conventional processing, depending on the fuel group.

None of the treatment options associated with the higher actinide targets or non-aluminum-clad fuels would produce transuranic waste.

4.1.1.4.3 Hazardous/Low-Level Mixed Waste

For this EIS analysis, DOE grouped hazardous and low-level mixed wastes together because none of the options is likely to produce significant quantities of either.

The highest hazardous/low-level mixed waste generation rates would be associated with Vitrification and Electrometallurgical Treatments, followed by Mechanical Dilution. However, due to the longer time required to process the loose uranium oxide in cans, the Materials Test Reactor-like fuels, and the highly enriched uranium/low enriched uranium (HEU/LEU) oxides and silicides requiring resizing or special packaging, conventional processing would produce the largest volume of hazardous or mixed waste for those fuel groups. Vitrification and Electrometallurgical Treatments generally would produce the next largest quantities (35 to 88 percent of that produced by conventional processing, depending on the fuel group). For the uranium and thorium metal fuels, Vitrification and Electrometallurgical Treatments produce the largest quantities of hazardous/low-level mixed waste, followed by conventional processing. For applicable fuel groups, the Direct Disposal/Direct Co-Disposal technology would consistently produce the smallest quantities of hazardous or mixed waste. The waste volumes that continued wet storage or the Melt and Dilute technology would produce would be roughly comparable and generally intermediate among the technologies. For the higher actinide targets, the two technologies being considered (Repackage and Prepare to Ship and Continued Wet Storage) would produce small, comparable quantities of hazardous or mixed waste.

When all applicable technologies are considered, conventional processing would generate

the largest volume (264 cubic meters) of hazardous and low-level mixed waste, which is less than 1 percent of the 30-year forecast.

4.1.1.4.4 Low-Level Waste

The Direct Disposal/Direct Co-Disposal and Repackage and Prepare to Ship technology options would produce the least low-level waste. The Mechanical Dilution and Melt and Dilute options would produce intermediate quantities of low-level waste, between 9 and 37 percent of the maximum volume generated and within approximately 150 percent of the minimum volume, depending on the fuel group. For applicable fuel groups, conventional processing would produce the most low-level waste. In each case, continued wet storage would produce the next highest volume due to the combined effect of storage time and generation rate. When all applicable fuel groups are included, conventional processing would generate 138,200 cubic meters of low-level waste (29 percent of the SRS low-level waste 30-year forecast) and continued wet storage would generate 56,650 cubic meters (12 percent of the forecast). Of the two options being considered for the higher actinide targets, the Repackage and Prepare to Ship option would produce the smallest quantity of low-level waste, 32 percent of that estimated for Continued Wet Storage.

4.1.1.4.5 By-products of converting SNF into a waste form that is suitable for disposal in a geologic repository

With the exception of continued wet storage under the No-Action Alternative, the technology options would convert the fuels into a waste form that is likely to be suitable for permanent disposal in a geologic repository. The radioactive inventory in the final waste form would be substantially greater than 99 percent of the original fuel inventory. Very small amounts of residual radioactivity would remain in secondary low-level, hazardous/mixed low-level, and transuranic waste streams as illustrated in Figures 4.1-1 through 4.1-7. SRS would use the surplus capacity in existing waste management facilities to treat, store, dispose of, or recycle the secondary waste in accordance with applicable regulations.

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The melt and dilute and vitrification technologies would release from the fuel matrix volatile fission products (primarily cesium) from the fuel matrix which would be recovered as illustrated in Figure 4.1-3 and Figure 4.1-5. Residual cesium, strontium, and plutonium from conventional processing (as well as volatile fission products from melt and dilute, and vitrification technology options) would be moved from the high-level waste tanks and separated into a high volume – low radioactivity salt stream and a low volume – high radioactivity slurry. The salt stream would be approximately 95 percent of the total (before separation) volume and the slurry would capture approximately 99.999 percent of the cesium, strontium, and plutonium activity (Choi 1992). The slurry would be encapsulated in glass and poured into canisters at the Defense Waste Processing Facility The canisters would be stored in a Glass Waste Storage Building for ultimate disposal in a geologic repository. The salt stream would be mixed into and solidified with concrete and disposed of in the Z-Area vaults.

TC

TC

4.1.1.4.6 Spent Fuel Canisters

DOE does not consider the SNF canisters resulting from alternate technology options to constitute a waste stream because they would be the end product of the new packaging options or new processing technology options being proposed. Nevertheless, the number of canisters is a useful measure of onsite storage space needed and the volume of the material that, after processing, could possibly be placed in a repository. Table 4.1-14 indicates the numbers of two types of canisters for the various technologies. The 17-inch canister would be used for co-disposal. The 24-inch canister would be used when the technology produces a vitrified product identical

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Table 4.1-14. Numbers of spent fuel co-disposal and high-level waste canisters.

Technology	Co-Disposal or Direct Disposal canisters	24-inch high-level waste canisters
Prepare for direct co-disposal	1,400	NA[a]
Repackage and prepare to ship	NA[b]	1
Melt and dilute	400	10
Mechanical dilution[c]	630	10
Vitrification technologies[d]	1,350	10
Electrometallurgical treatment	–	90
Conventional processing[e]	–	150
Continued wet storage	–	41

a. NA = not applicable, since DOE would use Co-Disposal.
b. Canisters would not be required to transfer material to another site.
c. Values were calculated for the press and dilute technology.
d. Values represent dissolve and vitrify and glass material oxidation and dissolution system technologies. The plasma arc technology would produce 490 canisters.
e. Values are for conventional processing the entire SNF inventory.

to the DPWF high-level waste borosilicate glass. After conventional processing, the 24-inch canisters would be stored in DWPF's Glass Waste Storage Building. The number of high-level waste canisters (Table 4.1-14) includes the secondary waste stream components generated by the technologies reported in Table 4.1-10.

4.1.1.5 Utility and Energy Resources

This section describes the estimated utility and energy requirements associated with each technology option under consideration in this EIS. Water, electricity, steam, and diesel fuel would be required to support many of the options. Estimates of water use include domestic water supplies and makeup water for process operations or equipment cooling. Steam is used primarily to heat facilities. Fuel consumption is based on use of diesel generators for backup power. Electrical requirements include that for normal office consumption such as heating, cooling, ventilation, and office equipment, and for specialized process-related equipment. The process equipment and the associated electrical demands would vary from option to option. All technologies would require canister loading and welding equipment. For the Melt and Dilute technology, the resistive heating associated with melting would require additional electricity. For aqueous processing, electrical requirements would include the operation of canyon pumps, circulators or mixers, and denitriting equipment. For Vitrification, electrical equipment would be used for resistive heating and dissolution. For Electrometallurgical Treatment, electricity would be used for resistive melting of fuels, operation of an electrolytic bath for metal purification, final melting of the refined uranium product, and blending down with depleted uranium.

Tables 4.1-15 through 4.1-18 list estimated utility and energy requirements for the technology options applicable to each fuel group. For each option, this analysis considered three handling phases as potential sources of energy consumption: wet storage (pretreatment storage), treatment, and dry storage (post-treatment storage pending final disposition). The durations for these phases are provided in Appendix E. The period and utility use rate associated with each phase would vary depending on the fuel group and the option. As discussed above, DOE summed utility use from each phase; the values listed in the tables represent the total projected utility use for each option in a given fuel group.

DOE used annual utility consumption rates to calculate the estimates in the tables (Bickford et

al. 1997); the rates are based on applicable cur-

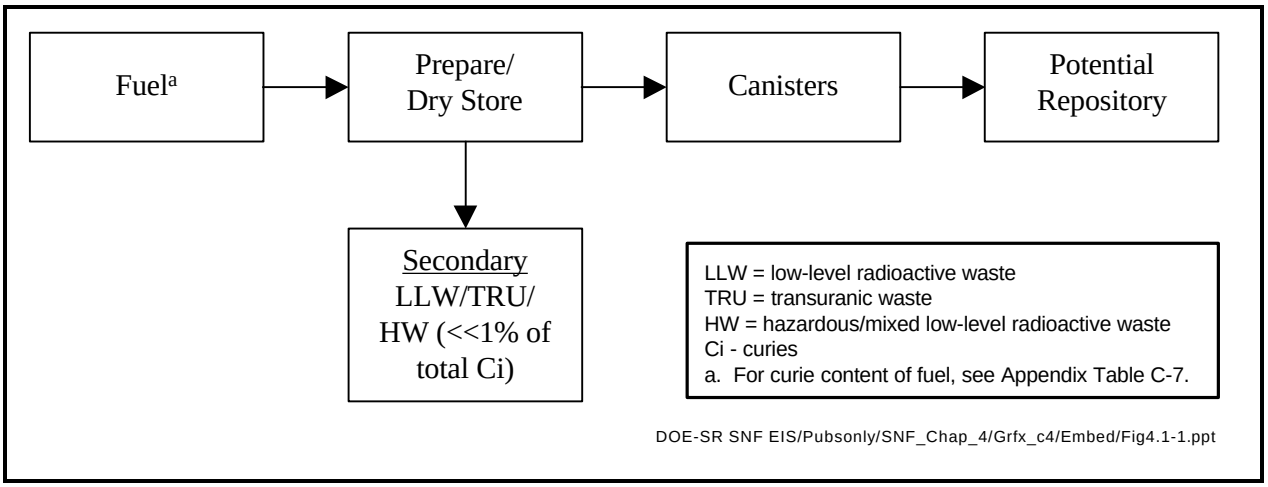

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Figure 4.1-1. Type and source of waste streams generated by the Prepare for Direct Co-Disposal technology option.

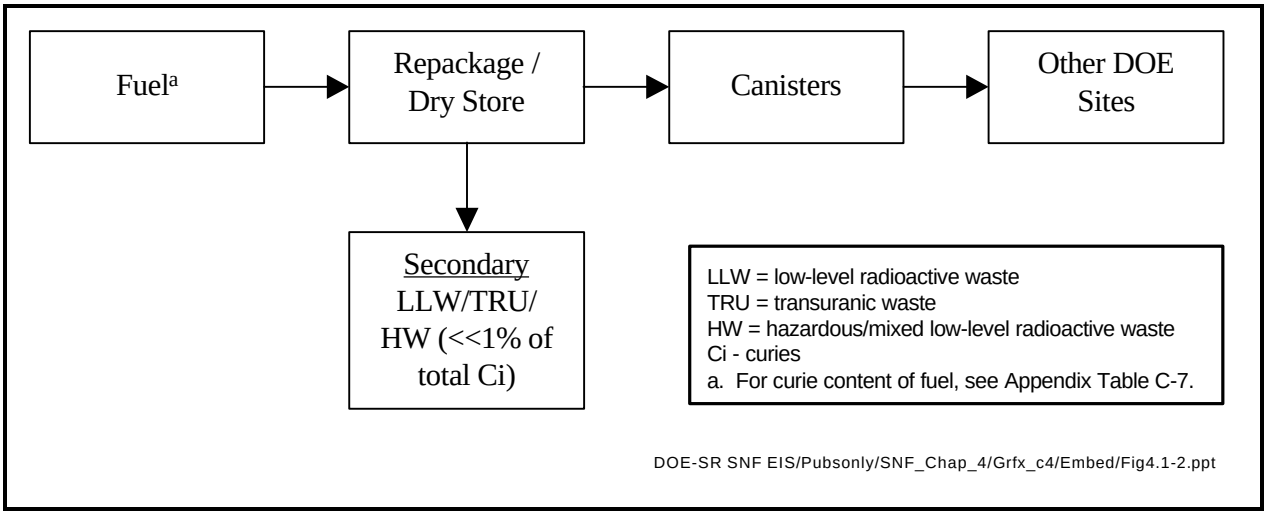

Figure 4.1-2. Type and source of waste streams generated by the Repackage and Prepare to Ship technology option.

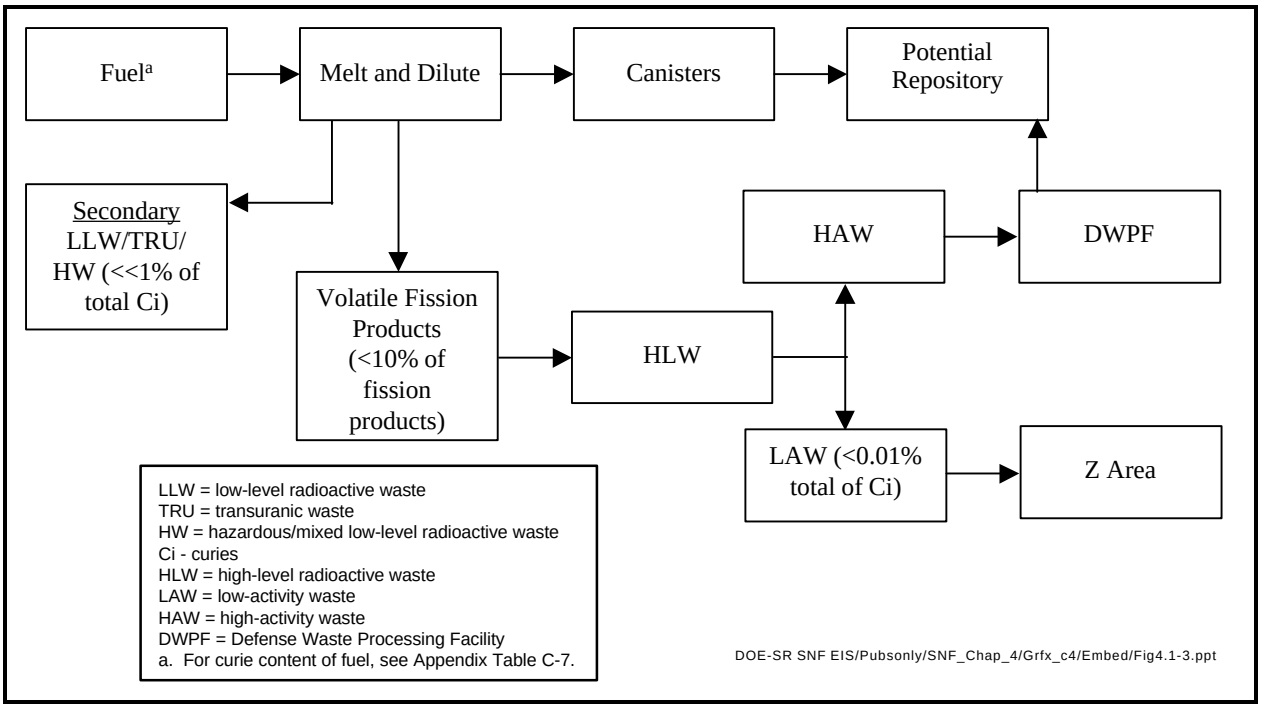

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Figure 4.1-3. Type and source of waste streams generated by the Melt and Dilute technology option.

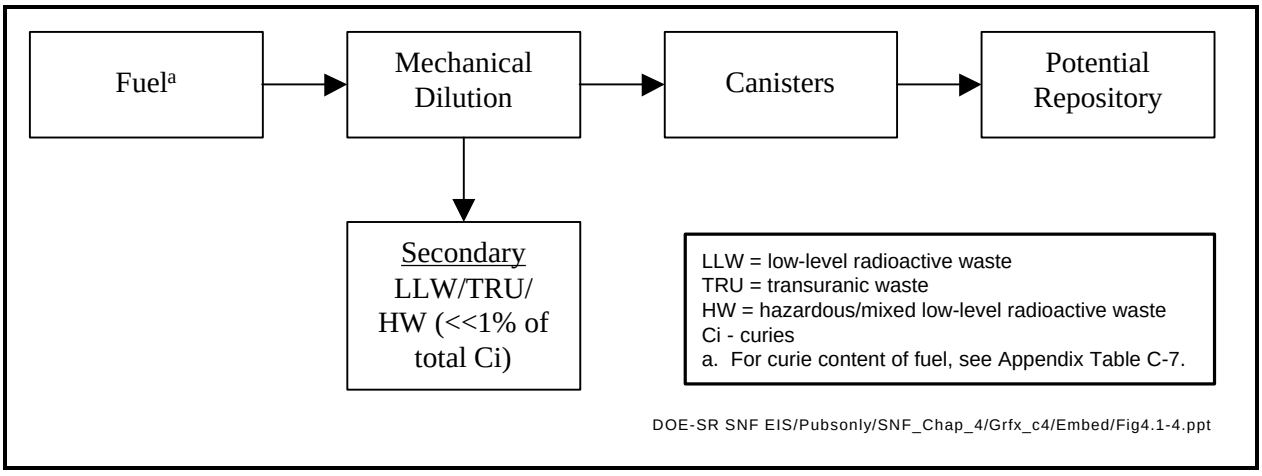

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Figure 4.1-4. Type and source of waste streams generated by the Mechanical Dilution technology option.

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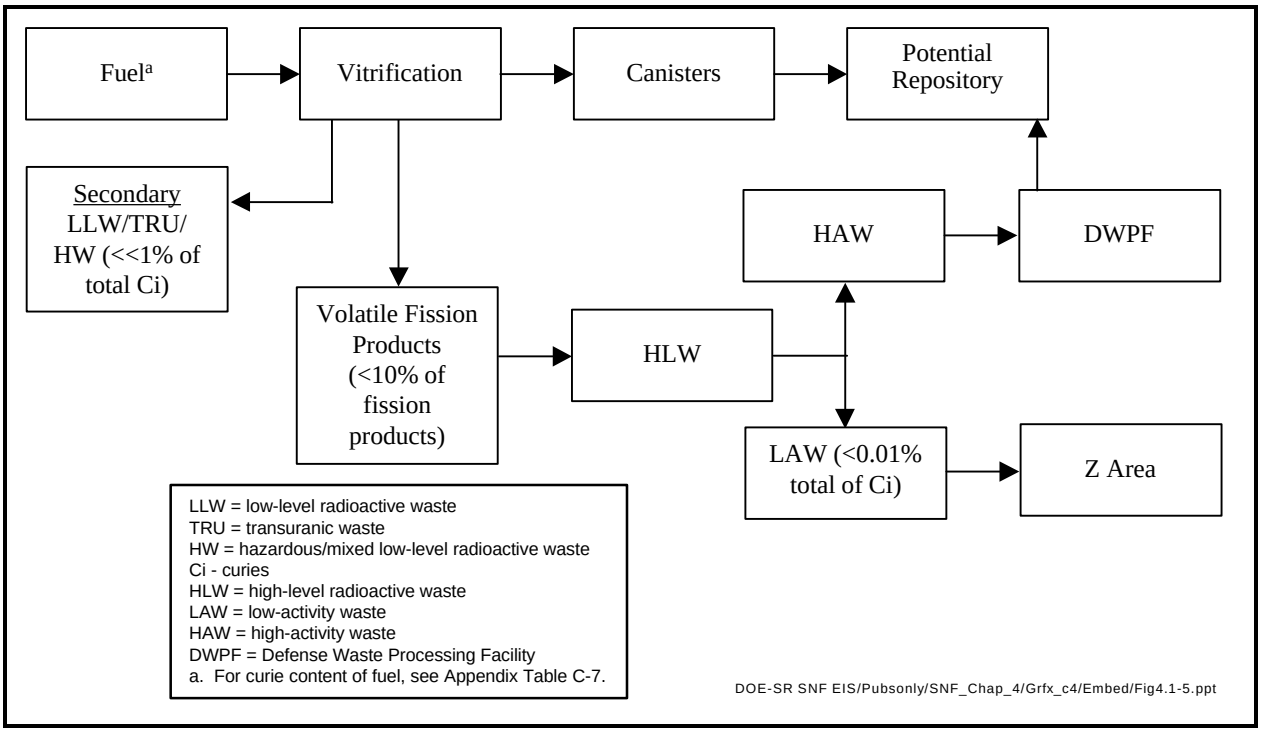

Figure 4.1-5. Type and source of waste streams generated by the Vitrification technology options.

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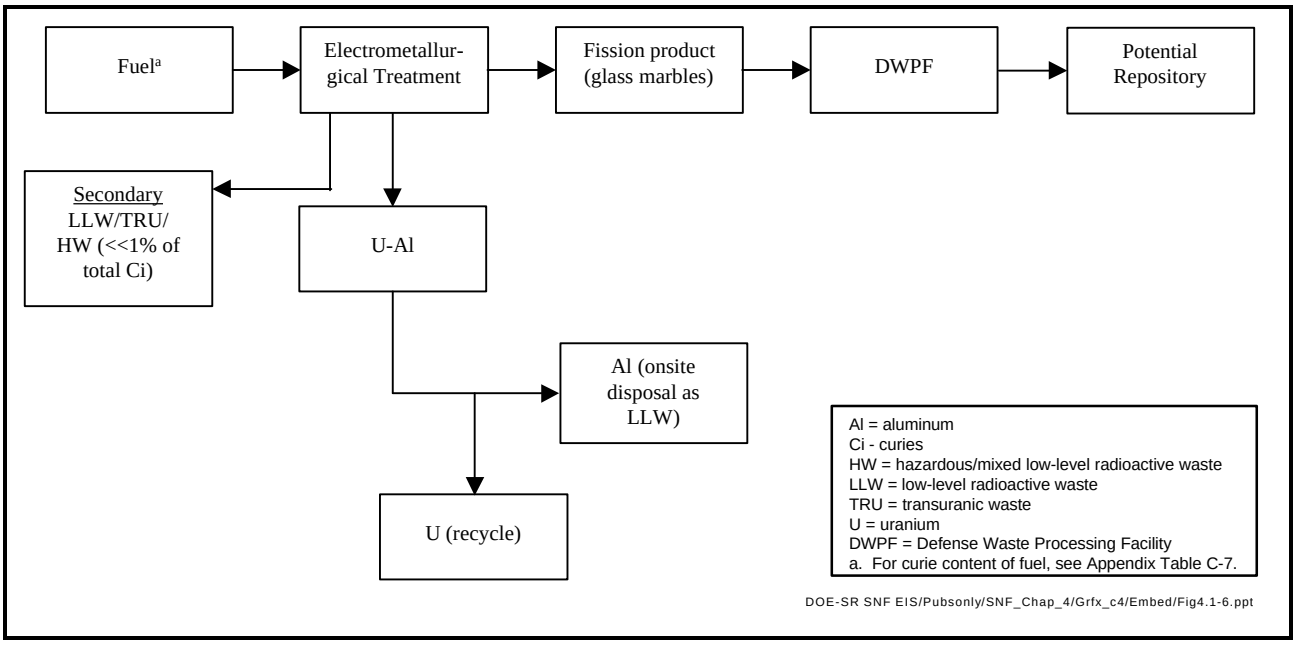

Figure 4.1-6. Type and source of waste streams generated by the Electrometallurgical Treatment technology option.

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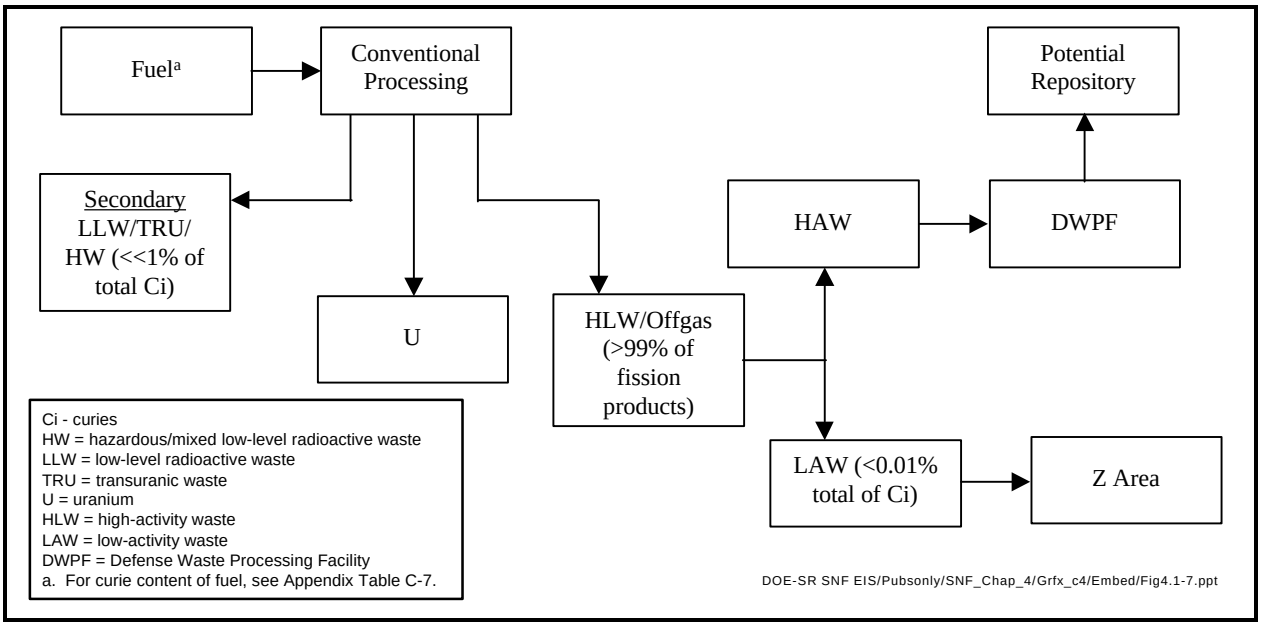

Figure 4.1-7. Type and source of waste streams generated by the Conventional Processing technology option.

rent and past SRS operations or on engineering judgments for new treatment technologies.

The following paragraphs describe estimated utility requirements for the options.

4.1.1.5.1 Water Use

Vitrification and Electrometallurgical Treatment would require the most water, followed by Conventional Processing. Total requirements for Vitrification and Electrometallurgical Treatment of all applicable fuel groups would be less than 6,000 liters per year, (the equivalent of 4.3 gallons per day) which is a minute portion (0.00001 percent) of groundwater withdrawal of more than 5×10^9 liters per year (DOE 1997). Due to the comparatively long period required to process the HEU/LEU oxides and silicides requiring resizing or special packaging (Fuel Group C) and the loose uranium oxide in cans (Fuel Group D), the Conventional Processing technology would require the greatest amount of water for those groups. For the higher actinide targets, Repackage and Prepare to Ship would require 67 percent of the water needed to support the only other option under consideration for that fuel group, Continued Wet Storage. In general, the Direct Disposal/Direct Co-Disposal, Melt and Dilute, Mechanical Dilution, and Repackage and Prepare to Ship technologies would require the least water for their applicable fuel groups, approximately 5 to 6 percent of the maximum requirement for a given group.

4.1.1.5.2 Electricity Use

Vitrification and Electrometallurgical Treatment would have the highest annual demand for electricity, followed by Conventional Processing. Differences in the time necessary to treat a fuel group under different options would affect total electricity requirements. Due to the longer period required to process the materials test reactor-like fuels (Fuel Group B), HEU/LEU oxides and silicides requiring resizing or special packaging (Fuel Group C), and loose uranium oxide in cans (Fuel Group D), Conventional Processing would require the most total electricity for those groups. For the higher actinide targets, Repackage and Prepare to Ship would require less than half the electricity needed to support continued wet storage. In general, for the appropriate fuel groups, the least electricity would be required to support Direct Co-Disposal and Mechanical Dilution.

Annually, the maximum impact alternative electrical demand is 23,600 megawatt-hours, which is approximately 3.5 percent of the current SRS annual usage of 660,000 megawatt-hours.

4.1.1.5.3 Steam Use

Where applicable, Conventional Processing would have the highest annual demand for steam. For higher actinide targets, Repackage and Prepare to Ship would require half the steam needed to support continued wet storage. In general, Direct Co-Disposal and Mechanical Dilution would require the least steam.

4.1.1.5.4 Diesel Fuel Use

For several options, DOE would use diesel fuel to support SNF treatment and storage. On an annual basis, Conventional Processing and Melt and Dilute would need the most diesel fuel. The least diesel fuel would be associated with the Vitrification and Electrometallurgical Treatment technologies, because both would require fuel only to support initial wet storage. The two options that DOE is considering for the higher actinide targets (Repackage and Prepare to Ship and Continued Wet Storage) would require comparable amounts of diesel fuel.

4.1.1.6 Environmental Justice

This section examines whether minority or low-income communities (as defined in Section 3.5.3) could receive disproportionately high and adverse human health and environmental impacts as a result of the actions described in this EIS. Even though DOE does not anticipate adverse health impacts from the options, it analyzed for the possibility of "disproportionately high and adverse human health or environmental effects on minority populations or low-income populations" (Executive Order 12898). Figures 3.5-1 and 3.5-2 show minority and low-income communities by census tract. This section discusses average radiation doses that individuals in those communities could receive and compares them to predicted doses that individuals in the other communities within the 80-kilometer- (50-mile) radius region could receive.

Figure 4.1-8 has SRS as the center of a circle with 22.5-degree sectors and concentric rings from 10 to 50 miles (16 to 80 kilometers) out from the center at 10-mile (16-kilometer) intervals. For this analysis, DOE calculated a fraction of the total population dose for each sector, laid the sector circle over the census tract map, and assigned each tract to a sector. If a tract fell in more than one sector, DOE assigned it to the sector with the largest dose value.

DOE analyzed impacts by comparing the per capita dose that each type of community would receive to doses other types of communities in the same ring would receive. To eliminate the possibility of diluting and masking impacts to a low-population community close to SRS with a high dose per person by including them with impacts to a high-population community farther from the Site, the analysis made comparisons in a series of concentric circles, the radii of which increase in 10-mile (16-kilometer) increments.

To determine the radiation dose received per person in each type of community, the analysis multiplied the number of people in each tract by that tract's dose value to obtain a total community population dose for each tract, summed these population doses in each concentric circle, and divided by the total community population in the circle to get a community per capita dose for each area of the circle. Because the per capita dose for communities (Table 4.1-19) would be constant for every alternative, the relative differences in impacts between communities would also be constant. Thus, Figure 4.1-9 and Table 4.1-19 indicate the distribution of per capita doses to types of communities in the 50-mile (80-kilometer) region. As shown in Figure 4.1-9, atmospheric releases would not disproportionately affect minority communities (population equal to or greater than 35 percent of the total population) or low income (equal to or greater than 25 percent of the total population) in the 50-mile region; that is, a comparison

Figure 4.1-8. Annular sectors around the Savannah River Site.

Table 4.1-19. Estimated per capita annual dose (rem) for identified communities in 80-kilometer (50-mile) region.[a]

	Low income		Minorities			
Distance	Less than 25 percent of population (rem)	Equal to or more than 25 percent of population (rem)	Less than 35 percent of population (rem)	35 percent to 50 percent of population (rem)	Equal to or more than 50 percent of population (rem)	All communities (rem)
0-10 miles (0-16 km[b])	1.1×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.2×10^{-5}	1.0×10^{-5}	1.1×10^{-5}
0-20 miles (0-32 km)	5.0×10^{-6}	5.0×10^{-6}	5.0×10^{-6}	7.0×10^{-6}	4.0×10^{-6}	5.0×10^{-6}
0-30 miles (0-48 km)	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	2.0×10^{-6}	3.0×10^{-6}
0-40 miles (0-64 km)	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	3.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}
0-50 miles (0-80 km)	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}

a. Per capita dose based on a population dose of 1 person-rem. Per capita doses for other population doses can be obtained by multiplying the values in this table by the population dose.
b. km = kilometers.

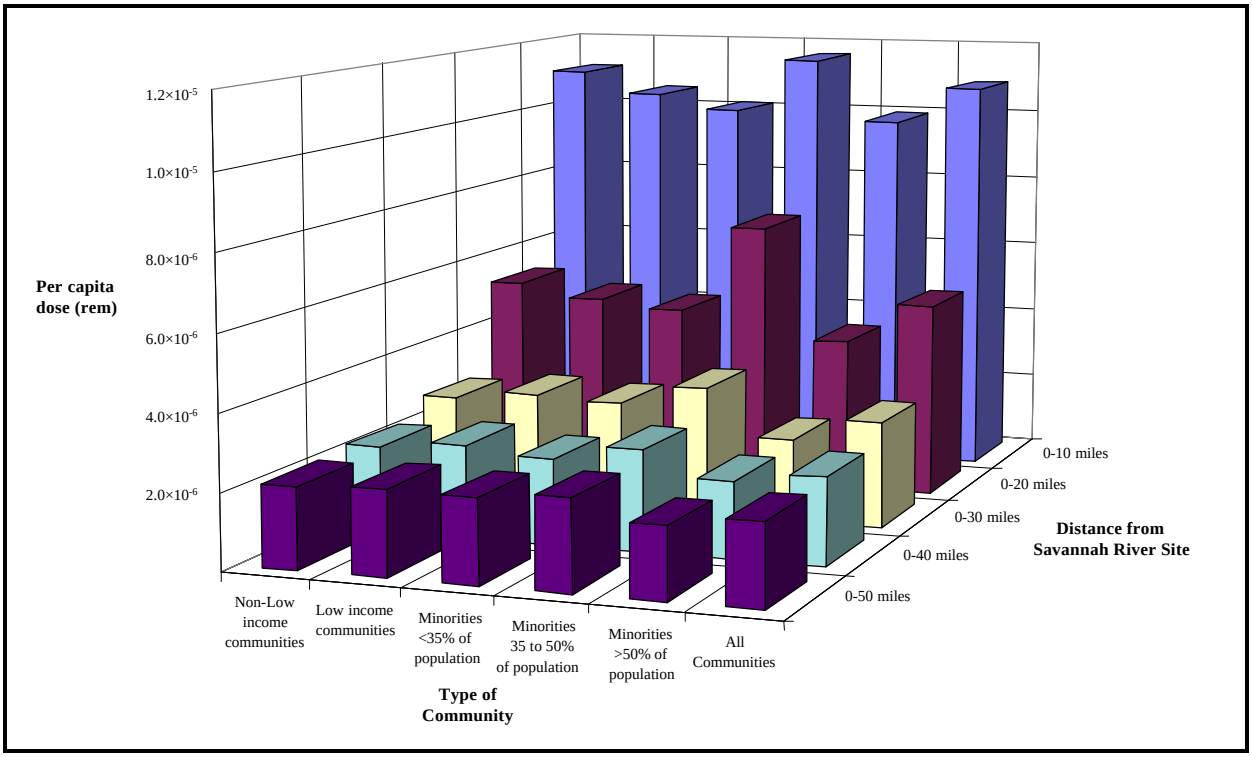

Figure 4.1-9. Distribution of a hypothetical unit population dose among SRS communities.

of per capita doses indicates that they do not vary greatly.

For example, DOE used an annual total population dose of 1 person-rem to prepare Figure 4.1-9 and its supporting data in Table 4.1-19. In comparison, the maximum annual total population dose of 0.56 person-rem for the maximum impact alternative (see Section 4.1.2) would result in 56 percent of the impact shown in Figure 4.1-9 and Table 4.1-19. For any other population dose, the per capita dose for communities can be determined by multiplying that population dose by the values listed in Table 4.1-19.

The distribution of carcinogenic and criteria pollutant emissions from routine operations and of criteria pollutants from construction activities would be essentially identical to those described for airborne radiological emissions because the distribution pathways would be the same. As a result, nonradiological emissions from any option would not cause disproportionate impacts on minority or low-income communities. Because nonradiological pollutant emissions would cause minimal impacts for any option, and because there would not be disproportionate distribution of these impacts among types of communities, environmental justice concerns would not be associated with the alternatives.

4.1.1.7 Transportation

This section discusses the potential radiological consequences of the onsite transportation of SNF and the potential consequences of transportation to a geologic repository. All onsite shipments (those that originate and terminate on SRS) would be by rail. Movements of SNF within an SRS area (e.g., H Area or F Area) are operational transfers, not onsite shipments. The potential consequences of shipping SNF from the SRS to a geologic repository are a conservative (based on worst-case number of shipments and mode of transportation) representation of impacts based on preliminary information. The full analysis of transportation impacts will be included in the EIS for a Geological Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada (currently in preparation).

4.1.1.7.1 Onsite Incident-Free Transportation Analysis [SRS]

The analysis assumed a crew of four engineers for each shipment and that the external dose rate 6.6 feet (2 meters) from the shipping cask was 100 millirem per hour (HNUS 1994a), 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 (39.4 feet [12 meters]). The duration of exposure would depend on the transport vehicle speed. In addition, vehicle crew time would depend on the distance of each shipment.

Table 4.1-20 summarizes the collective doses (person-rem) and health effects (latent cancer fatalities) associated with a single incident-free onsite shipment of SNF at SRS.

To determine the incident-free transportation dose for management of all SRS spent nuclear fuel, it is necessary to calculate the total dose over all shipments. DOE has estimated that it would take approximately 150 rail shipments to de-inventory the Receiving Basin for Offsite Fuels to the L-Area Disassembly Basin. This action would occur under all alternatives, including the No-Action Alternative. The radiation dose to the crew from these shipments is estimated to be approximately 0.57 person-rem, which could result in 2.3×10^{-4} latent cancer fatalities.

DOE has estimated that it would take approximately 300 rail shipments to transport the contents of the L-Area Disassembly Basin (including the fuel that was previously in the Receiving Basin for Offsite Fuels) to the Transfer and Storage Facility; the Transfer, Storage, and Treatment Facility; or the F- and H-Area Canyons. This action would occur under all alternatives, except the No-Action Alternative. Assuming the bounding location for the

Table 4.1-20. Collective doses and health effects for onsite incident-free SNF shipments.[a]

Shipment origin/destination	Crew dose per shipment (person-rem)	Number of LCFs[b] per shipment
		Crew
L Area/H Area	3.80×10^{-3}	1.52×10^{-6}
L Area/F Area	4.10×10^{-3}	1.64×10^{-6}
F Area/H Area	1.40×10^{-3}	5.60×10^{-7}
P Area/H Area	4.90×10^{-3}	1.96×10^{-6}
P Area/F Area	3.88×10^{-3}	1.55×10^{-6}
C Area/H Area	3.33×10^{-3}	1.33×10^{-6}
C Area/F Area	4.20×10^{-3}	1.68×10^{-6}

a. Derived from HNUS (1994a).
b. LCF = latent cancer fatality.

Transfer and Storage Facility or the Transfer, Storage, and Treatment Facility, the radiation dose to the crew from these shipments is estimated to be approximately 1.23 person-rem which could result in $4.9\text{x}10^{-4}$ latent cancer fatalities. Therefore, for the No-Action Alternative, the total radiation dose to the shipping crew would be approximately 0.57 person-rem, which could result in $2.3\text{x}10^{-4}$ latent cancer fatalities. For all other alternatives, the total radiation dose to the crew would be approximately 1.8 person-rem, which could result in $7.2\text{x}10^{-4}$ latent cancer fatalities.

4.1.1.7.2 Incident-Free Transportation Analysis [Geologic Repository]

DOE estimated the impacts of shipping SNF from SRS to a theoretical geologic repository in the Western United States (approximately 4,000 kilometers [2,500 miles] from SRS) by truck. TC This analysis assumes all shipments from SRS, approximately 1,400 (worst case among the alternatives), would be by truck because the impacts would bound the impacts of rail shipments. Because the transport of SRS spent fuel would use existing highways, it would represent a very small fraction of national highway traffic. Consequently, there would be negligible impacts on land use; air quality; hydrology; biological resources and cultural resources; socioeconomics; noise; aesthetics; utilities, energy, and materials; or waste management. The analysis of the potential impacts of transporting SRS spent nuclear fuel to the repository focuses on the potential radiological impacts to workers and the public.

DOE recognizes that it cannot predict with any certainty the specific routes that would be used to ship SNF to a repository. Nonetheless, the analysis uses current regulations governing highway shipments to select actual highway routes to estimate the potential environmental impacts of national transportation. TC Assumed distances within the various rural, suburban, and urban population zones can be found on Table 4.1-21.

Loading Operations

Prior to shipping the fuel, DOE would load it into NRC certified Type B shipping casks. The potential dose to involved workers from the loading operation would be less than that expected at a commercial nuclear facility because the radionuclide inventory of commercial fuel is higher than that of the DOE SNF. The dose would be further limited by worker rotation and other administrative controls. DOE expects any dose to uninvolved workers would be negligible because they would not have tasks that could result in radiation exposure. Likewise, DOE expects radiation exposure to the public would not occur because of the distance of the loading operations from the areas of public access.

Table 4.1-21. Incident-free radiological impacts of 1,400 offsite truck shipments of spent nuclear fuel to the proposed Yucca Mountain Geologic Repository.

Exposure group	Unit risk factors (person-rem kilometer)[a]			Kilometers traveled		
	Rural	Suburban	Urban	Rural	Suburban	Urban
Occupational	4.6×10^{-5}	1.0×10^{-4}	1.7×10^{-4}	3,292.6	570.2	65.9
Off-link[b]	1.2×10^{-7}	1.6×10^{-5}	1.1×10^{-4}	3,292.6	570.2	65.9
On-link[c]	5.0×10^{-6}	1.5×10^{-5}	1.5×10^{-4}	3,292.6	570.2	65.9
Stops	1.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}	3,292.6	570.2	65.9

	Collective dose (person-rem)			Total collective dose	LCF[d]
	Rural	Suburban	Urban		
Occupational	212	80	16	308	0.123
General population					
Off-link[b]	1	13	10	24	0.012
On-link[c]	23	12	14	49	0.024
Stops	553	96	11	660	0.330
General population total					0.366

a. The methodology, equations, and data used to develop the unit risk factors are discussed in Madsen et al. (1986) and Neuhauser and Kanipe (1992). Cashwell et al. (1986) contains a detailed explanation of the use of unit risk factors.
b. Off-link general population are persons within 800 meters (2,625 feet) of the highway.
c. On-link general population are persons sharing the highway.
d. LCF = latent cancer fatality.

Transportation to a Geologic Repository

To estimate the potential impacts of incident-free transportation of SNF to a repository, the analysis considered both the public and workers. Unit risk factors commonly used in a number of other DOE EISs were used to determine the potential person-rem exposure per kilometer for both workers and public. In the case of the general population, both off-link and on-link doses were calculated. The off-link dose could affect persons within 800 meters (2,625 feet) of the highway; the on-link dose could affect persons sharing the highway. Table 4.1-21 presents the potential incident-free radiological impacts from 1,400 shipments of SNF from the SRS to a theoretical geologic repository. As can be seen from the table, potential latent cancer fatalities could result in less than 1 additional death from radiation over the life of the shipments.

4.1.1.7.3 Onsite Transportation Accident Analysis [SRS]

DOE analyzed radiological impacts from potential accidents to the onsite maximally exposed individual from onsite rail shipments. The analysis calculated doses using the RADTRAN computer code (Neuhauser and Kanipe 1992) with site-specific meteorology, and calculated risk using site-specific rail accident rates and accident probabilities (HNUS 1994b).

The analysis assumed a release of the maximum reasonably foreseeable amount of radioactive material for the type of SNF shipped on SRS (HNUS 1994b). Radiological doses were modeled for three human receptor groups: the onsite worker population, members of the public residing near SRS, and the maximally exposed offsite individual. The consequences are expressed as excess latent cancer fatalities in each receptor group.

Table 4.1-22 summarizes the radiation doses resulting from the most severe reasonably foreseeable onsite transportation accident and associated latent cancer fatalities.

4.1.1.7.4 Transportation Accident Analysis [Geologic Repository]

EC Potential impacts from accidents resulting from transporting SNF to a geologic repository are not quantified in this document but have been analyzed in the EIS for a Geologic Repository for Disposal of Spent Nuclear Fuel and High-Level EC Radioactive Waste at Yucca Mountain, Nye County, Nevada. Previous EISs, including the Foreign Research Reactor Spent Fuel EIS (DOE 1996) and the Programmatic Spent Fuel EIS (DOE 1995b) analyzed the potential accident impacts of transporting SNF. The following discussions summarize the types of accidents that could be expected. Impacts are presented in Table 4.1-23.

Loading Operation

In general, accidents from loading operations could be caused by unplanned contact (bumping) during lifting or handling of casks, canisters, or fuel assemblies. Initiating events could include fires, explosions, earthquakes, cask tor nadoes, canister or basket drops, and loaded shipping drops. The Interim Management of Nuclear Materials at SRS EIS (DOE 1995a) assessed the radiological impacts from potential accidents associated with preparing, storing, and onsite shipment of some spent nuclear fuel.

Transportation to a Geologic Repository

Several types of accidents potentially could occur while transporting SNF. The first type of accident, resulting in the most radiological exposure to the public, assumes the breach of a shipping cask during an accident resulting in the release of a fraction of its contents to the air. This accident would be very unlikely. The second type of accident would involve truck wrecks that could result in non-radiological fatalities to workers or members of the public. The probability of an accident is dependent upon the number of shipments made and total miles traveled. EC

4.1.2 IMPACTS OF THE ALTERNATIVES

As discussed in Chapter 2, none of the options for the management of SNF, except Continued Wet Storage, would address the requirements of all six fuel types. Therefore, DOE must consider combinations of technologies to satisfy the purpose and need identified in Chapter 1. This

Table 4.1-22. Impacts on SRS workers, maximally exposed offsite individuals, and offsite population from SNF transportation accidents on Savannah River Site.

Accident frequency	Worker dose (rem)	Probability of a worker LCF[b]	MEI[c] dose (rem)	Probability of a LCF to the MEI	Population dose (person-rem)	Population LCFs
1.28×10^{-4}	2.78	1.11×10^{-3}	2.2×10^{-5}	1.08×10^{-8}	0.16	8.21×10^{-5}

a. Source: DOE (1995a).
b. LCF = latent cancer fatality.
TC c. MEI = maximally exposed individual.

Table 4.1-23. Truck transportation accident analysis impacts.

Radiological impacts				Traffic impacts		
Risk factor (person-rem/ shipment)[a]	Maximum number shipments	Total (person-rem)	Total LCFs	Risk factor (fatality/ shipment)[b]	Maximum number shipments	Total fatality
TC 1.79×10^{-5}	1,400	0.025	1.25×10^{-5}	1.12×10^{-4}	1,400	0.16

LCF = latent cancer fatalities.
a. DOE (1996).
TC b. Adapted from DOE (1999).

section provides the results of analyzing combinations of the technology options applicable to the fuel groups. Excluding continued wet storage, there are more than 700 combinations of technology options and fuel groups that could be analyzed. However, it would be impractical and unreasonable to do so. DOE has identified four sets of combinations for analysis as alternatives in this EIS (in addition to No Action) which it believes are representative. These four alternatives are the Minimum Impact Alternative, Direct Disposal Alternative, Preferred Alternative, and Maximum Impact Alternative. The data in Section 4.1.1 can be used to compile the impacts of other configurations of viable cases.

Continued wet storage for all fuel types is the No-Action Alternative. National Environmental Policy Act (NEPA) regulations require the evaluation of No Action, (which would not meet the purpose and need described in Chapter 1); however, it provides a baseline against which DOE can compare the action alternative combinations.

The second alternative, Minimum Impact, would result in the smallest environmental impacts to human health. It is also the environmentally-preferred alternative.

The third alternative is Direct Disposal. All fuel types that could be dry stored would be. Higher Actinide Targets and Non-Aluminum-Clad Fuels would be Repackaged and Prepared to Ship Offsite. Uranium and Thorium Metal Fuels and Loose Uranium Oxide in Cans would undergo conventional processing.

The fourth alternative is the Preferred Alternative. Melt and Dilute would be used to treat the Materials Test Reactor-like fuels, most of the HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (Group C), and most of the Loose Uranium Oxide in Cans (Group D). Group A and the remaining Group C and Group D fuels (<10 percent of the material in these fuel groups) would be treated with conventional processing. Finally, the Higher Actinide Targets and the Non-Aluminum-Clad fuels would be Repackaged and Prepared to Ship offsite.

EC

The final alternative would apply the chemical processing option to all the fuel except the higher actinide targets and non-aluminum-clad SNF and probably would produce the greatest environmental impacts, and therefore, provides an upper bound. It is termed the Maximum Impact Alternative. Section 2.4 provides a complete description of the SNF management alternatives.

Tables 4.1-24 through 4.1-26 list the impacts of the five alternatives summed from the operational impacts of each appropriate technology presented in Section 4.1.1. The following sections describe the alternatives and the bases for their selection. The conclusions from Section 4.1.1.5 on environmental justice would apply to all the alternatives.

DOE based the values listed for annual radiation dose to the noninvolved worker, the offsite maximally exposed individual, and the 620,000-person population surrounding SRS on the sum of the annual doses for each technology-fuel group included in the alternative. Since the time intervals over which these annual doses would occur might not coincide, this method could overestimate the annual doses that actually would occur.

The values in Table 4.1-26 for health effects to the noninvolved worker, maximally exposed individual, and the offsite population for the No-Action Alternative represent current reactor area emissions (including two SNF wet basins) for the entire period of analysis. The values for the other alternatives would be incremental above these baseline values. Summing these baseline and incremental values would be conservative, however, because there would not be two SNF wet basins operating over the entire 38-year period of analysis.

Table 4.1-24. Estimated maximum incremental concentrations of nonradiological air pollutants for the noninvolved worker.

Pollutant	Averaging Time	Regulatory Standard[a]	No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative
Toxic Pollutants (mg/m^3)							
Nitric acid	24-hour	5	0.03	0.02	2.75	2.62	7.95
1,1,1-Trichloroethane	24-hour	1,900	–	–	0.02	0.02	0.05
Benzene	24-hour	3.19	–	–	0.02	0.02	0.05
Ethanolamine	24-hour	6	0.03	0.02	0.02	0.02	0.03
Ethyl benzene	24-hour	435	–	–	0.01	0.01	0.02
Ethylene glycol	24-hour	None	0.03	0.02	0.02	0.02	0.03
Formaldehyde	24-hour	0.75	0.03	0.02	0.02	0.02	0.03
Glycol ethers	24-hour	80	0.03	0.02	0.02	0.02	0.03
Hexachloronaphthalene	24-hour	0.2	0.03	0.02	0.02	0.02	0.03
Hexane	24-hour	1,800	0.03	0.02	0.03	0.03	0.06
Manganese	24-hour	5	–	–	0.01	0.01	0.02
Mercury	24-hour	0.1	–	–	0.01	0.01	0.02
Methyl alcohol	24-hour	260	0.03	0.02	0.02	0.02	0.03
Methyl ethyl ketone	24-hour	590	0.03	0.02	0.02	0.02	0.03
Methyl isobutyl ketone	24-hour	410	–	–	0.01	0.01	0.02
Methylene chloride	24-hour	86.7	–	–	0.02	0.02	0.05
Napthalene	24-hour	50	0.03	0.02	0.02	0.02	0.03
Phenol	24-hour	19	–	–	0.01	0.01	0.02
Phosphorus	24-hour	0.1	–	–	0.01	0.01	0.02
Sodium hydroxide	24-hour	2.0	–	–	0.01	0.01	0.02
Toluene	24-hour	754	0.03	0.02	0.03	0.03	0.06
Trichloroethene	24-hour	537	–	–	0.01	0.01	0.02
Vinyl acetate	24-hour	None	–	–	0.01	0.01	0.02
Xylene	24-hour	435	0.03	0.02	0.05	0.05	0.10
Criteria Pollutants ($\mu g/m^3$)							
Nitrogen oxides	Annual	NA	–	0.05	38.2	36.4	111
Total Suspended Particulates (total dust)	8-hour	15	–	0.02	0.35	0.34	0.99
Particulate Matter (<10 μm)	8-hour	5	–	0.09	0.08	0.08	0.05
	24-hour	NA	–	0.99	0.86	0.87	0.62
Carbon monoxide	8-hour	55	0.03	0.25	1.81	1.82	4.78
	1-hour	NA	0.03	0.79	5.65	5.68	14.93
Sulfur dioxide	Annual	NA	–	0.02	0.04	0.04	0.08
	8-hour	13	–	0.02	0.31	0.30	0.86
	3-hour	NA	–	0.02	0.72	0.70	2.07
Gaseous fluorides	1-month	None	–	-	0.10	0.10	0.29
	1-week	NA	–	-	0.18	0.17	0.52
	24-hour	NA	–	-	0.55	0.52	1.59
	12-hour	NA	–	-	0.80	0.76	2.32
Ozone (as VOC)	1-hour	0.2	–	nc	nc	nc	nc

– = no air emission associated with this combination.
NA = not applicable.
nc = not calculated.
VOC = volatile organic compound.
a. 29 CFR 1910.1000, Subpart Z and OSHA 8-hour time-weighted averages.

Table 4.1-25. Estimated maximum incremental concentrations of nonradiological air pollutants at the Site boundary.

Pollutant	Averaging Time	Regulatory Standard[a]	No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative	
Toxic Pollutants (mg/m^3)								
Nitric acid	24-hour	125	–	–	0.11	0.10	0.31	
1,1,1-Trichloroethane	24-hour	9,550	0.03	0.03	0.03	0.03	0.03	
Benzene	24-hour	150	–	–	0.01	0.01	0.02	
Ethanolamine	24-hour	200	0.03	0.03	0.03	0.03	0.03	
Ethyl benzene	24-hour	4,350	–	–	0.01	0.01	0.02	
Ethylene glycol	24-hour	650	0.03	0.03	0.03	0.03	0.03	
Formaldehyde	24-hour	15	0.03	0.03	0.03	0.03	0.03	
Glycol ethers	24-hour	+	0.03	0.03	0.03	0.03	0.03	
Hexachloronaphthalene	24-hour	1	0.03	0.03	0.03	0.03	0.03	
Hexane	24-hour	200	0.03	0.03	0.03	0.03	0.03	TC
Manganese	24-hour	25	–	–	0.01	0.01	0.02	
Mercury	24-hour	0.25	–	–	0.01	0.01	0.02	
Methyl alcohol	24-hour	1,310	0.03	0.03	0.03	0.03	0.03	
Methyl ethyl ketone	24-hour	14,750	0.03	0.03	0.03	0.03	0.03	
Methyl isobutyl ketone	24-hour	2,050	–	–	0.01	0.01	0.02	
Methylene chloride	24-hour	8,750	–	–	0.01	0.01	0.02	
Napthalene	24-hour	1,250	0.03	0.03	0.03	0.03	0.03	
Phenol	24-hour	190	–	–	0.01	0.01	0.02	
Phosphorus	24-hour	0.5	–	–	0.01	0.01	0.02	
Sodium hydroxide	24-hour	20	–	–	0.01	0.01	0.02	TC
Toluene	24-hour	2,000	0.03	0.03	0.03	0.03	0.03	
Trichloroethene	24-hour	6,750	–	–	0.01	0.01	0.02	
Vinyl acetate	24-hour	176	–	–	0.01	0.01	0.02	
Xylene	24-hour	4,350	0.03	0.03	0.03	0.03	0.03	
Criteria Pollutants ($\mu g/m^3$)								
Nitrogen oxide	Annual	100	0.03	0.02	1.17	1.12	3.36	
Total Suspended Particulates	Annual	75	0.03	0.02	0.02	0.02	0.02	
Particulate Matter (<10 µm)	Annual	50	–	–	0.01	0.01	0.02	
	24-hour	150	–	–	0.05	0.04	0.13	
Carbon monoxide	8-hours	10,000	0.03	0.07	0.49	0.50	1.31	
	1-hour	40,000	0.03	0.37	3.60	3.57	9.76	
Sulfur dioxide	Annual	80	–	0.02	0.02	0.02	0.02	
	24-hour	365	–	0.03	0.07	0.07	0.13	
	3-hour	1300	–	–	0.34	0.32	0.98	
Gaseous fluoride	1-month	0.8	–	–	0.01	0.01	0.02	
	1-week	1.6	–	–	0.02	0.01	0.04	
	24-hour	2.9	–	–	0.03	0.02	0.07	
	12-hour	3.7	–	–	0.05	0.04	0.13	
Ozone (as VOC)	1-hour	235	–	0.16	0.38	0.41	0.80	

– = no air emission associated with this option.
+ = no state standard.
VOC = volatile organic compound.
a. SCDHEC standard No. 2 (criteria pollutants) and No. 8 (toxic pollutants). TC

Table 4.1-26. Impacts from alternatives.[a]

Impact	No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative[b]	Maximum Impact Alternative
Health Effects for the Entire Period of Analysis (1998-2035)[f]					
MEI[c] dose (millirem)	0.63[d]	6.1×10^{-4}	7.2×10^{-3}	0.19	0.67
MEI LCF[e] probability	3.1×10^{-7}[d]	3.0×10^{-10}	3.6×10^{-9}	9.5×10^{-8}	3.4×10^{-7}
Population dose (person-rem)	22.6[d]	0.022	0.077	6.9	8.7
Population LCFs (unitless)	0.011[d]	1.1×10^{-5}	3.8×10^{-5}	3.4×10^{-3}	4.4×10^{-3}
Collective worker dose (person-rem)	760	690	840	841	2,100
Collective worker LCFs (unitless)	0.30	0.28	0.34	0.33	0.84
Noninvolved worker dose (millirem)	4.25[d]	5.0×10^{-3}	0.02	1.53	1.53
Noninvolved worker LCF probability	1.7×10^{-6}[d]	2.0×10^{-9}	9.6×10^{-9}	6.1×10^{-7}	6.3×10^{-7}
Annual Radiological Air Emission Impacts					
Maximum annual MEI[d] dose (millirem)	0.02[d]	6.1×10^{-4}	7.4×10^{-4}	0.044	0.015
Maximum annual population dose (person-rem)	0.59[d]	0.022	0.027	1.6	0.56
Maximum annual noninvolved worker dose (millirem)	0.11[d]	5.0×10^{-3}	6.0×10^{-3}	0.36	0.12
Annual Radiological Liquid Emission Impacts					
Maximum annual MEI dose (millirem)	0	0	1.4×10^{-3}	4.2×10^{-5}	0.057
Maximum annual population dose (person-rem)	0	0	4.9×10^{-3}	2.4×10^{-4}	0.19
Waste Generation (cubic meters) for the Entire Period of Analysis (1998-2035)					
High-level waste					
Liquid	2,300	660	1,200	1,050	10,500
Equivalent DWPF canisters	38	11	20	17	160
Saltstone	6,100	1,800	3,200	2,700	27,000
Transuranic waste	0	15	360	563	3,700
Hazardous/low-level mixed waste	76	25	46	103	267
Low-level waste	57,000	20,000	31,000	35,260	140,000
Utilities and Energy Required for the Entire Period of Analysis (1998-2035)					
Water (millions of liters)	1,100	660	1,400	1186	8,000
Electricity (megawatt-hours)	46,000	27,000	81,000	116,000	600,000
Steam (millions of kilograms)	340	195	520	650	3,600
Diesel fuel (thousands of liters)	230	180	2,300	2760	22,000

a. In the event that fuel receipts are less than those reported in Chapter 1, the values in this table that report impacts over the entire period of analysis would be less. Instructions for scaling impacts are provided in the appropriate Chapter 4 tables that provide input to this table.
b. In the calculation of preferred alternative impacts, all the HEU/LEU oxides and silicides requiring resizing or special packaging have been accounted for in the melt and dilute technology even though a very small percentage would be conventionally processed. On the other hand, the loose-uranium-oxide-in-cans preferred alternative impacts do consider that 60 percent would be conventionally processed and the remaining 40 percent would be melted and diluted.
c. MEI = maximally exposed offsite individual.
d. Reflects current reactor-area emissions (including two SNF wet basins).
e. LCF = latent cancer fatality.
f. To calculate an annual impact, divide a number by 38. To calculate an impact for a given duration, multiply the annual impact by the duration in years. For example, the annual dose to the MEI from the preferred alternative would be 0.005 mrem (0.17/38). The estimated dose to the MEI until a storage facility would be operational (18 years from now) would be 0.040 mrem (0.005x8).

4.1.2.1 No-Action Alternative

Under the No-Action Alternative, SRS would continue to receive shipments of SNF from foreign research reactors, domestic research reactors, and other DOE sites. DOE would store the fuel in the L-Reactor Disassembly Basin or the Receiving Basin for Offsite Fuels, in addition to the currently stored SNF, under continued wet storage, and would ship the non-aluminum-clad fuel from these basins offsite. DOE would maintain the wet storage basins, performing upgrades as necessary to maintain proper water quality. The continued long-term underwater storage of aluminum-based SNF could lead to increased corrosion with increased environmental, health, and safety vulnerabilities. The No-Action Alternative consists of cases A8, B8, C8, D8, E8, and F8 (Table 4.1-27).

4.1.2.2 Minimum Impact Alternative

The identification of the Minimum Impact Alternative required both quantitative and quantitative analyses. The first step identified the minimum-impact technology for each fuel group for each analytical parameter (e.g., volume of high-level waste, air concentrations). However, the selection process often resulted in a combination of high and low impacts among parameters for a specific fuel group-technology combination cases; in other words, no clearly identified "best" or "worst" configuration was identified. Therefore, the second step was a qualitative examination of trends in configurations of cases that identified overall minimum impacts. Human health effects and environmental pollution impacts received slightly greater weight than consumption of natural resources or waste disposal space. In addition, impacts to the general public received slightly greater weight than those to SRS workers. The analysis indicates that cases A1, B1, C1, D3, E2, and F2 would provide minimum impacts (Table 4.1-28). Although other analysts could select different cases, DOE believes that the range of impacts from reasonable choices of minimum-impact scenarios would be small and that the impacts of this combination would be representative of the lower bound of impacts from the proposed action.

4.1.2.3 Direct Disposal Alternative

This alternative combines the New Packaging and the Conventional Processing Technologies. Materials Test Reactor-like fuels and HEU/LEU Oxides and Silicides (except the failed and sectioned fuels) would be treated using the Direct Disposal/Direct Co-Disposal technology and placed in the Transfer and Storage Facility with a minimum of treatment (e.g., cold-vacuum drying and canning). The repackaging of the higher actinide targets and non-aluminum-clad fuels in the Transfer and Storage Facility would use the Repackage and Prepare to Ship technology. The uranium and thorium metal fuel, loose uranium oxide in cans, and failed and sectioned fuel from the HEU/LEU Oxides and Silicides fuel group would be treated using the Conventional Processing Alternative to alleviate the potential health and safety vulnerabilities discussed in Section 2.4.3.2 and because this material probably would not be suitable for placement in a geologic repository if treated with the Direct Disposal/Co-Disposal option. Therefore, the Direct Disposal alternative consists of cases A7, B1, C1, D7, E2, and F2 (Table 4.1-29). TC

4.1.2.4 Preferred Alternative

DOE proposes to implement several of the technologies identified in Section 2.2 to manage spent nuclear fuel at SRS. These technologies are Melt and Dilute, Conventional Processing, and Repackage and Prepare to Ship. Each of these technologies would treat specific groups of spent nuclear fuel, as described below. The technology and fuel group combinations form DOE's Preferred Alternative in this EIS. The configuration of this preferred alternative is identified in Table 4.1-30.

4.1.2.4.1 Melt And Dilute

DOE has identified the Melt and Dilute process as the preferred method of treating most (about 97 percent by volume or about 32,000 MTRE) of the aluminum-based SNF considered in this EIS. DOE will continue to pursue a research and development program leading to a demonstration of the technology in FY 2001 using full-size irradiated research reactor spent nuclear fuel assemblies. With a successful demonstration of the technology, DOE expects to have ready a treatment facility to perform production melt and dilute operations in FY 2008. DOE will ensure the continued availability of SRS conventional processing facilities until we have successfully demonstrated implementation of the Melt and Dilute treatment technology.

The fuel proposed for the preferred Melt and Dilute technology includes the Material Test Reactor-like fuel, most of the Loose Uranium Oxide in Cans fuel, and most of the HEU/LEU Oxide and Silicide fuel. Exceptions are the uranium and thorium fuel, failed and sectioned oxide and silicide fuel, some loose uranium oxide in cans fuel, the Higher Actinide Targets, and non-aluminum-clad fuel.

If DOE identifies any health or safety concerns involving any aluminum-based SNF prior to the melt and dilute facility becoming operational, DOE could use F and H Canyons to stabilize the material of concern, if the canyons were not decommissioned.

4.1.2.4.2 Conventional Processing

DOE has identified conventional processing to manage a relatively small volume of aluminum-based SNF at the SRS (about 3 percent by volume; less than 3,000 MTRE) that presents a potential health and safety vulnerability or is in a form that may be unacceptable for placement in a geologic repository. That SNF includes the Experimental Breeder Reactor-II fuel, the Sodium Reactor Experiment fuel, the Mark-42 targets and the core filter block from the Uranium and Thorium Metal fuel group; the failed or sectioned Tower Shielding Reactor, High Flux Isotope Reactor, Oak Ridge Reactor, and Heavy Water Components Test Reactor fuels and a Mark-14 target from the HEU/LEU Oxides and Silicides fuel group; and the Sterling Forest Oxide (and any other powdered/oxide fuel that may be received at SRS while H Canyon is still in operation) from the Loose Uranium Oxide in Cans fuel group.

4.1.2.4.3 Repackaging

DOE proposes to repackage the non-aluminum-clad fuel at SRS and transfer the material to dry storage. DOE would transfer the non-aluminum-clad fuel to that facility for storage pending offsite shipment. DOE expects transfer operations would begin in time to support closing the Receiving Basin for Offsite Fuels by 2007. Depending on receipt schedules for research reactor fuels and the operating schedule for the melt and dilute facility, DOE could deinventory the Receiving Basin for Offsite Fuels and move any remain fuel to the Building 105-L wet basin prior to packaging the fuel for dry storage.

The Preferred Alternative would include cases A7, B3, C3, D3, E2, and F2 (Table 4.1-30).

4.1.2.4.4 Continued Wet Storage

DOE proposed to maintain the higher actinide target fuel group in continued wet storage pending decisions on final dispositon.

4.1.2.5 Maximum Impact Alternative

This alternative provides the upper bound on the range of impacts from potential configurations. It would provide conventional processing for all SNF except the higher actinide targets and the non-aluminum-clad fuels selected for offsite shipment and deemed inappropriate for conventional processing. The higher actinide targets would be repackaged for potential offsite shipment and dry-stored until DOE made a decision regarding their disposition. The non-aluminum-clad fuels would be packaged for shipment and dry stored until they were ready for shipment to the Idaho National Engineering and Environmental Laboratory.

Analyses of the maximum impact alternative are conservative in that they assume that the entire SNF inventory would be processed in the canyons, which would produce the greatest impacts of all the treatment options. No credit is taken for discontinuing use of the canyons and processing some of the inventory in a new treatment facility. The Conventional Processing Alternative would include cases A7, B7, C7, D7, E2, and F2 (Table 4.1-31). DOE believes that this combination would provide an upper bound on impacts.

4.2 Accident Analysis

This section summarizes risks to the public and workers from potential accidents associated with the technology options for SNF management at the SRS.

An accident is a sequence of one or more unplanned events with potential outcomes that endanger the health and safety of workers and the public. An accident can involve a combined release of energy and hazardous materials (radiological or chemical) that might cause prompt or latent health effects. The sequence usually begins with an initiating event, such as a human error followed by an explosion, or an earthquake followed by structural failure. A succession of other events, such as a ventilation system failure, that are dependent or independent of the initial event, could affect the magnitude of the accident and the materials released. Initiating events fall into three categories:

- *Internal initiators* normally originate in and around the facility but are always a result of facility operations (equipment or structural failures, human errors, internal flooding).

- *External initiators* are independent of facility operations and normally originate outside the facility (aircraft crashes, nearby explosions, and toxic chemical releases at nearby facilities that affect worker performance); some can affect the ability of the facility to maintain confinement of hazardous materials because of structural damage.

- *Natural* phenomena initiators are natural occurrences that are independent of facility operations and of *events* at nearby facilities or operations (earthquakes, high winds, floods, lightning, snow). Natural phenomena initiators could affect external facilities, which could in turn affect other facilities and compound the progression of the accident.

Table 4.2-1 summarizes the estimated impacts to workers and the public from potential accidents for each SNF technology option. All the options would require the use of the Receiving Basin for Offsite Fuels and the L-Reactor Disassembly Basin. All except Continued Wet Storage would require the construction and operation of a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility.

The table lists the impacts of potential accidents in relation to the phases required to implement each option. They list only the accident with the worst impacts based on the maximally exposed offsite individual. Appendix D contains details of the impacts of other postulated accidents. Table 4.2-1 lists potential accident consequences as latent cancer fatalities, without consideration of the accident's probability. The calculation of latent cancer fatalities from population dose is performed in the same manner as for non-accident radiological health effects presented in section 4.1.1.3.1. TC

DOE estimated impacts to three receptors: (1) an uninvolved worker 2,100 feet (640 meters) from the accident location as discussed in DOE (1994), (2) the maximally exposed individual at the SRS boundary, and (3) the offsite population in an area within 50 miles (80 kilometers).

Many of the analysis results presented in Table 4.2-1 are substantially different from those given in the draft EIS. DOE has continued to conduct research and development, including accident analyses, to determine the feasibility of implementing technologies and the potential health and safety consequences of doing so. In some cases design changes have been TC

Table 4.2-1. Estimated maximum consequence accident for each technology.

Option	Accident Frequency	Consequences			
		Noninvolved Worker (rem)	MEI (rem)	Offsite Population (person-rem)	Latent Cancer Fatalities
Continued Wet Storage (No Action)[a]					
RBOF (high wind-induced criticality)	Once in 26,000 years	13	0.22	12,000	6.2
L-Reactor basin (basin-water draindown)	Once in 500 years	0.014	0.016	(b)	(b)
Direct Co-Disposal					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Repackage and Prepare to Ship					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Conventional Processing					
Processing phase in F/H Canyons (coil and tube failure)	Once in 14,000 years	13	1.3	78,000	39
Melt and Dilute					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Melt and dilute phase (earthquake induced spill with loss of ventilation)	Once in 200,000 years	30	0.5	21,000	10
Mechanical Dilution					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Mechanical dilution phase (criticality with loss of ventilation)	Once in 33,000 years	0.71	0.074	3,000	1.5
Vitrification Technologies					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Vitrification phase (earthquake-induced release with loss of ventilation)	Once in 200,000 years	0.10	0.0017	71	0.035
Electrometallurgical Treatment					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Electrometallurgical phase (metal melter earthquake induced spill with loss of ventilation)	Once in 200,000 years	30	0.5	21,000	10

MEI = Maximally Exposed Individual.
RBOF = Receiving Basin for Offsite Fuels.
a. All alternatives would use RBOF and the L-Reactor Disassembly Basin; therefore, accidents in these facilities are possible for each technology.
b. Not available.

considered specifically to reduce the potential for accidents with adverse consequences. During that process, assumptions about the design and operation of the proposed technologies have changed. Changes in the assumptions have resulted in changes in the outcome of the accident analyses. Details concerning the analyses are found in Appendix D of this EIS.

TC

For all of the accidents, there is a potential for injury or death to involved workers in the vicinity of the accident. In some cases, the impacts to the involved worker would be greater than to the noninvolved worker. However, prediction of latent potential health effects becomes increasingly difficult to quantify as the distance between the accident location and the receptor decreases because the individual worker exposure cannot be precisely defined with respect to the presence of shielding and other protective features. The worker also may be acutely.injured or killed by physical effects of the accident itself. DOE identified potential accidents through a detailed hazard assessment and estimated impacts using the AXAIRQ computer model (Simpkins 1995a,b), as discussed in Appendix D.

Results of accident calculations listed in Table 4.2-1 have been updated since the Draft EIS to incorporate evolution of the technology alternatives and to incorporate information that was not available at the time the Draft EIS was prepared.

TC

4.3 Construction Impacts

This section describes environmental impacts that could result from construction activities associated with SNF management at SRS. These activities would include the construction of a Transfer and Storage Facility under theNew Packaging Technology or the construction of a Transfer, Storage, and Treatment Facility under the New Processing Technology or Conventional Processing. DOE does not expect such construction activities to have appreciable impacts on geologic resources, groundwater, traffic, transportation, or cultural resources, as explained below

4.3.1 GEOLOGY AND GROUNDWATER

DOE would confine the construction of new facilities to previously disturbed and developed areas and, therefore, expects little or no environmental impacts to the geologic resources of the area. Neither the construction nor the operation of the proposed Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would affect groundwater in the area. The proposed DOE action to remove stored fuels from existing basins would eliminate a potential source of environmental releases (leaks from wet basins). The Transfer and Storage Facility or Transfer, Storage, and Treatment Facility could include the capability to perform wet receipt and unloading of SNF.

4.3.2 TRAFFIC AND TRANSPORTATION

DOE would transport construction materials, wastes, and excavated materials associated with building the proposed facilities both on and off SRS. These activities would result in increases in the operation of personal vehicles by construction workers, commercial truck traffic, and traffic associated with the daily operations of SRS. However, increases in worker and materials traffic would be small in comparison to existing traffic loads. Increased traffic congestion would be minimal.

4.3.3 CULTURAL RESOURCES

As discussed in Section 3.6, activities associated with the proposed action and alternatives for SNF management at SRS that could affect cultural resources would be the use of the three candidate sites for the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility. These sites are in reactor areas (L, C, and P) within 100 to 400 yards (91 to 366 meters) of the reactor buildings. The Savannah River Archaeological Research Program has not examined these sites. The Site Use Program, which requires a permit for clearing land on the SRS, usually initiates archaeological investigations. DOE would direct an investigation of the selected site before starting facility design and construc-

tion. Although there were homesites at or near the proposed facility sites in C and L Areas, the likelihood of historic resources surviving the construction of the reactors in the early 1950s, before the enactment of regulations to protect such resources would be small (Sassaman 1997).

The potential for the presence of prehistoric sites in the candidate locations also is limited. The L-Area site is in archaeological site density Zone 3, which has the least potential for prehistoric sites of significance. The C-Area site is in Zones 2 and 3 and has more potential. Zone 2 includes areas of moderate archaeological site density. The P-Area site is in Zone 2. However, as with any historic sites, reactor construction activities probably destroyed or severely damaged prehistoric deposits. DOE would direct an examination of the selected location for prehistoric resources before starting the design and construction of the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility (Sassaman 1997).

4.3.4 SURFACE WATER RESOURCES

Construction at SRS must comply with the requirements of South Carolina stormwater management and sediment reduction regulations, which became effective in 1992 as part of the Clean Water Act. These regulations and their associated permits require DOE to prepare erosion and sediment control plans for all projects, regardless of the land area. Runoff from the construction site would be part of a stormwater management and sedimentation control plan to minimize potential discharges of silts, solids, and other contaminants to surface-water streams. Effective January 2, 1997, the South Carolina Department of Health and Environmental Control (SCDHEC) approved General Permit coverage for stormwater management and sediment reduction at the SRS (SCDHEC 1996). Although the General Permit does not exempt any land-disturbing and construction activities from the requirements of State stormwater management and sediment control regulations, it does preclude the necessity of SCDHEC plan review and approval for land disturbing and construction activities at the SRS.

Before beginning construction, DOE would develop erosion and sediment control plans for the planned facilities. After construction and depending on the location of the construction site, the *SRS Stormwater Pollution Prevention Plan* (WSRC 1993), which is a requirement of the general NPDES stormwater permit covering industrial activities (Permit SCR000000), would include applicable erosion and sediment control measures; inclusion in the plan would not be necessary if the facility to be constructed was in the drainage area of a stormwater collection system permitted as part of NPDES Permit SC0000175.

4.3.5 AIR RESOURCES

The potential construction of facilities for the management of SNF would cause emissions of fugitive dust (particulate matter) from land-clearing activities and exhaust emissions from construction equipment (earth-moving vehicles, diesel generators). DOE has considered such impacts for activities at SRS that were similar in facility size and application and concluded that impacts to air quality would be minimal (DOE 1995a,b) and would have no effect on SRS compliance with state and Federal ambient air quality standards. Concentrations of pollutants emitted during construction activities would be at least an order of magnitude less than the South Carolina ambient air quality standards.

4.3.6 ECOLOGICAL RESOURCES

DOE is considering three brown field sites for the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility, if they are not constructed in a renovated reactor: C Area, L Area, and P Area. As noted in Section 3.4, the sites would encompass approximately 60,700 square meters (15 acres), including the main building and land required for ancillary facilities. The Treatment Facility could also be constructed on a previously disturbed site inside the F-Area or H-Area fences.

All construction activity for the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would take place within the

boundary of one of the three reactor areas in an already-developed brownfield area. Undeveloped portions of the three proposed sites provide some low-quality wildlife habitat.

Construction of the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would involve the movement of workers and construction equipment, and would be associated with relatively loud noises from earth-moving equipment, portable generators, pile-driving equipment, pneumatic tools, drills, hammers, and the like. Although noise levels in construction areas could be as high as 110 dBA, these high local noise levels would not extend far beyond the boundaries of the project site.

Table 4.3-1 gives the attenuation of construction noise over relatively short distances. At 120 meters (400 feet) from the construction site, construction noises would range from approximately 60 to 80 dBA. Golden et al. (1980) suggest that noise levels higher than 80 to 85 dBA are sufficient to startle or frighten birds and small mammals. Thus, there would be minimal Potential for disturbing birds and small mammals outside a 120-meter radius from the construction site.

Although noise levels would be relatively low outside the immediate area of construction, the combination of construction noise and human activity probably would displace small numbers of animals (e.g., songbirds and small mammals) that could forage, feed, nest, rest, or den in the area. Construction-related disturbances are likely to create impacts to wildlife that would be small, temporary (approximately 24 months), and localized. Some animals could be driven from the area permanently, while others could become accustomed to the increased noise and activity and return to the area. Species likely to be affected (e.g., gray squirrel, opossum, white-tailed deer) are common to ubiquitous in these areas. Construction would not disturb any threatened or endangered species, would not degrade any critical or sensitive habitat, and would not affect any jurisdictional wetlands.

Table 4.3-1. Peak and attenuated noise (in dBA) levels expected from operation of construction equipment.[a]

Source	Noise level (peak)	Distance from source			
		50 feet[b]	100 feet	200 feet	400 feet
Heavy trucks	95	84-89	78-83	72-77	66-71
Dump trucks	108	88	82	76	70
Concrete mixer	105	85	79	73	67
Jackhammer	108	88	82	76	70
Scraper	93	80-89	74-82	68-77	60-71
Dozer	107	87-102	81-96	75-90	69-84
Generator	96	76	70	64	58
Crane	104	75-88	69-82	63-76	55-70
Loader	104	73-86	67-80	61-74	55-68
Grader	108	88-91	82-85	76-79	70-73
Dragline	105	85	79	73	67
Pile driver	105	95	89	83	77
Fork lift	100	95	89	83	77

a. Source: Golden et al. (1980).
b. To convert feet to meters, multiply by 0.3048.

4.3.7 IMPACTS FROM RENOVATING AN EXISTING FACILITY

4.3.7.1 Waste Generation

As discussed in Section 2.3.2.3, DOE could locate the Transfer, Storage, and Treatment Facility in a renovated reactor area, such as the 105-L facility. This would require decontamination and removal of components and systems and subsequent construction activities inside the reactor building and would result in impacts that would not occur during the construction of a virgin facility. Impacts would include generation of radioactive waste during decontamination, removal and construction. DOE has estimated that decontamination and removal and construction activities would result in the generation of approximately 476 m^3 of low-level waste over the total duration of the activities (WSRC 1998). Eventual decontamination and decommissioning (D&D) of the Transfer, Storage, and Treatment Facility (either stand-alone or in a renovated reactor facility) also would result in generation of radioactive waste.

4.3.7.2 Worker Health

DOE could locate the Transfer, Storage, and Treatment Facility in a renovated reactor area, such as the 105-L facility. This would require decontamination and removal of components and systems and subsequent construction activities inside the reactor building and would result in impacts that would not occur during the construction of a virgin facility. Impacts would include radiation exposure of workers performing these activities. The decontamination and removal and construction activities would result in a total collective worker radiation dose of 32 person-rem, based on 54 total workers and a duration of 1 year to complete all activities (Nathen 1998). The collective worker dose is estimated to result in 1.3×10^{-3} latent cancer fatalities. Eventual decontamination and decommissioning (D&D) of the Transfer, Storage, and Treatment Facility (either stand-alone or in a renovated reactor facility) also would result in radiation exposure of D&D workers.

4.3.8 SOCIOECONOMIC IMPACTS

The implementation of the alternatives discussed in this EIS could result in the construction and operation of a Transfer and Storage Facility or a Transfer, Storage and Treatment Facility, which could in turn cause incremental socioeconomic impacts in the SRS area Section 2.3.2 discusses the construction and operation of the Transfer and Storage Facility. Its construction would cost an estimated $200 million. A 2-year construction period would result in a short-term increase of fewer than 500 jobs in the region, approximately 75 percent of which would be in construction. This would be an increase in consruction jobs of approximately 2 percent (from about 16,000) and an increase of considerably less than 1 percent in total employment for the region (REMI 1995). After the 2-year period, employment would return back to its previous equilibrium. The small temporary increases in employment would not present significant impacts to the regional economy, services, or infrastructure. TC

DOE would construct the treatment phase of the Transfer, Storage, and Treatment Facility after the Transfer and Storage phase was constructed; the construction periods would not overlap. The treatment phase would require less effort to construct and would employ fewer construction employees.

None of these construction activities would significantly increase regional employment or population, and socioeconomic impacts would be negligible.

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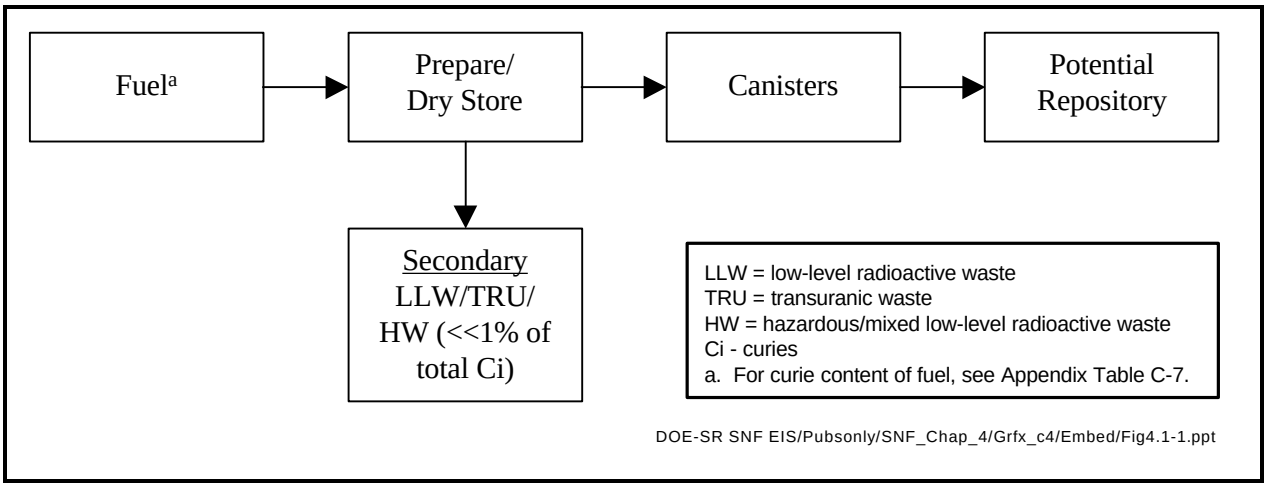

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Figure 4.1-1. Type and source of waste streams generated by the Prepare for Direct Co-Disposal technology option.

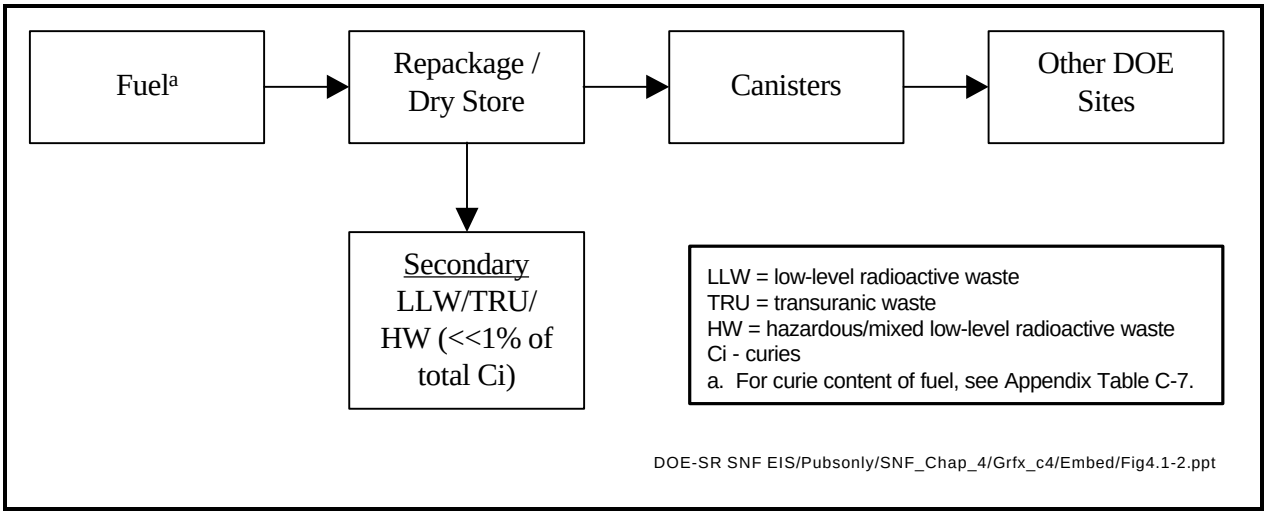

Figure 4.1-2. Type and source of waste streams generated by the Repackage and Prepare to Ship technology option.

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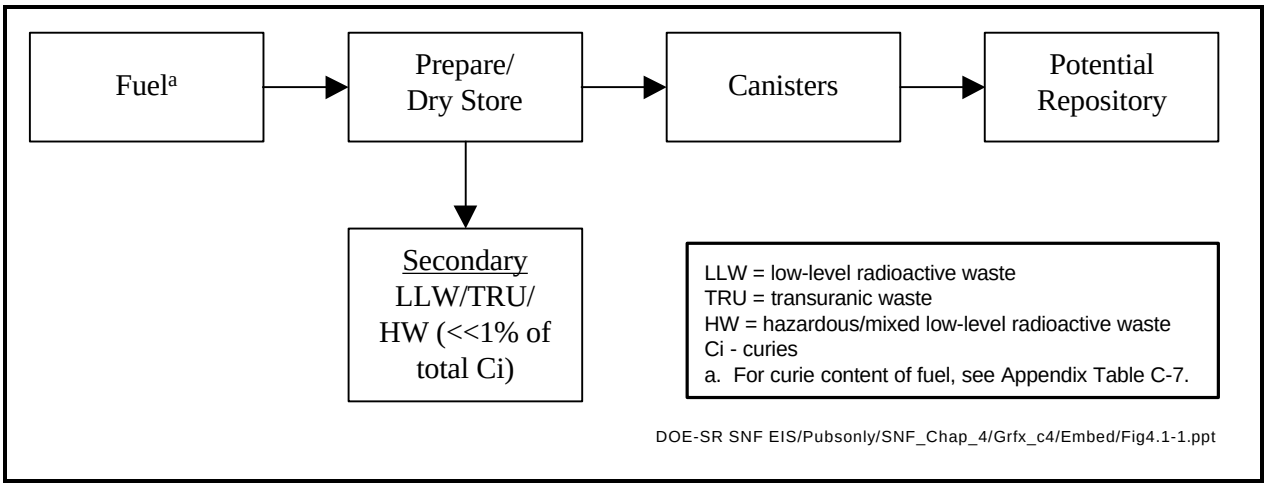

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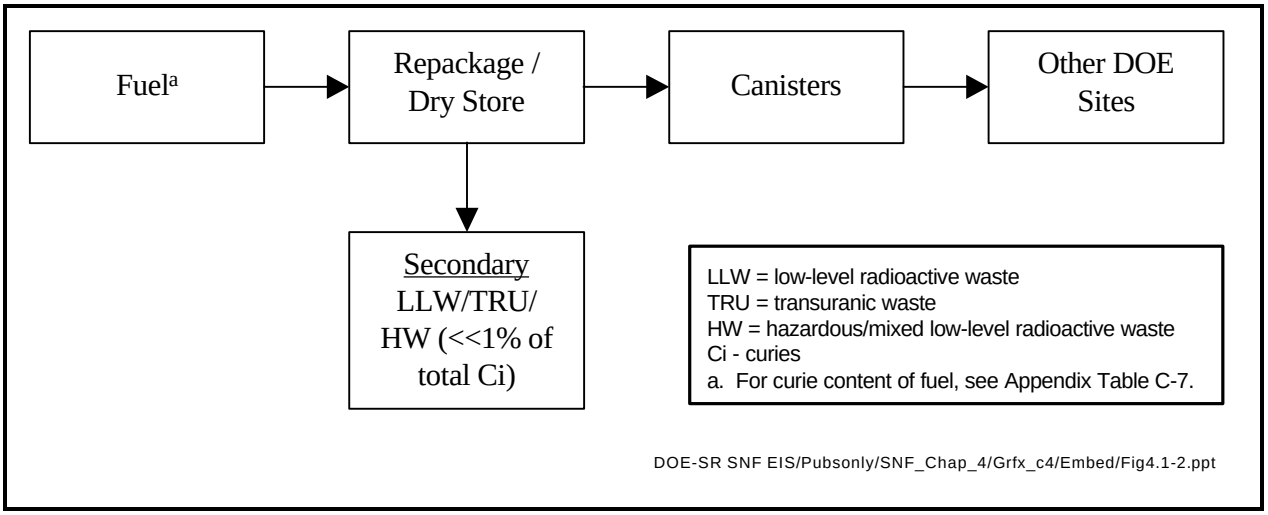

Figure 4.1-2. Type and source of waste streams generated by the Repackage and Prepare to Ship technology option.

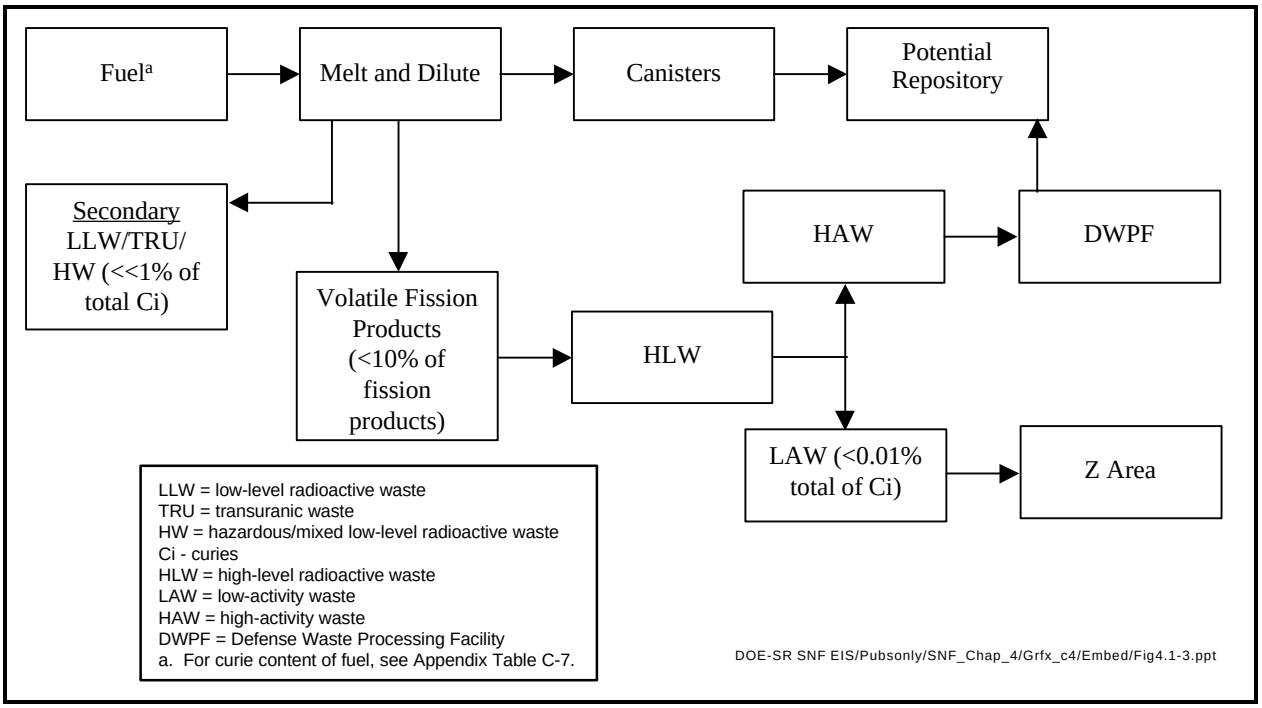

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Figure 4.1-3. Type and source of waste streams generated by the Melt and Dilute technology option.

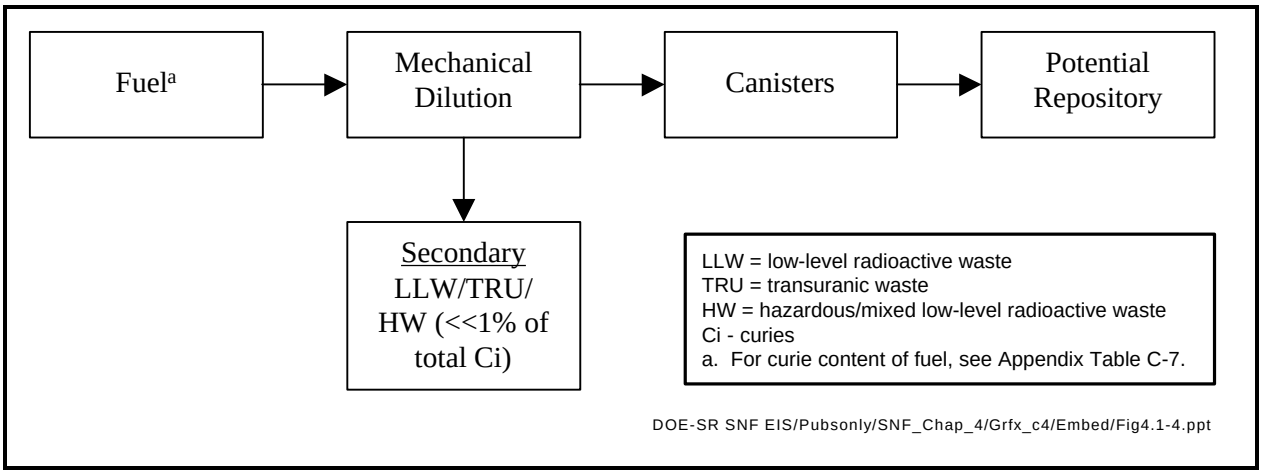

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Figure 4.1-4. Type and source of waste streams generated by the Mechanical Dilution technology option.

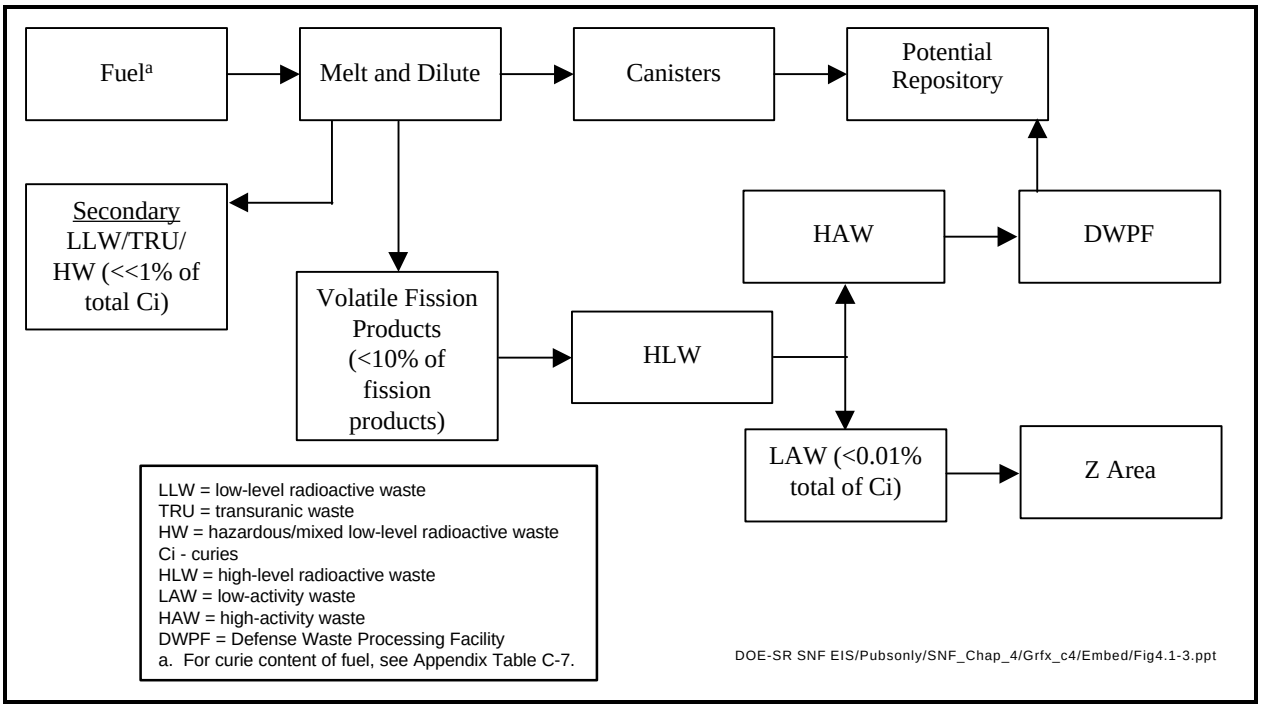

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Figure 4.1-3. Type and source of waste streams generated by the Melt and Dilute technology option.

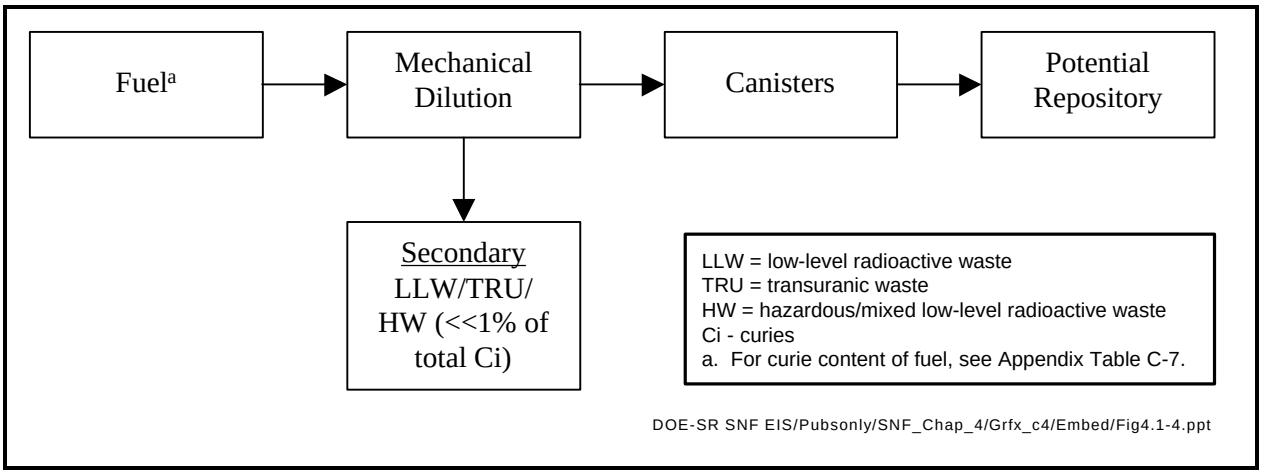

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Figure 4.1-4. Type and source of waste streams generated by the Mechanical Dilution technology option.

EC

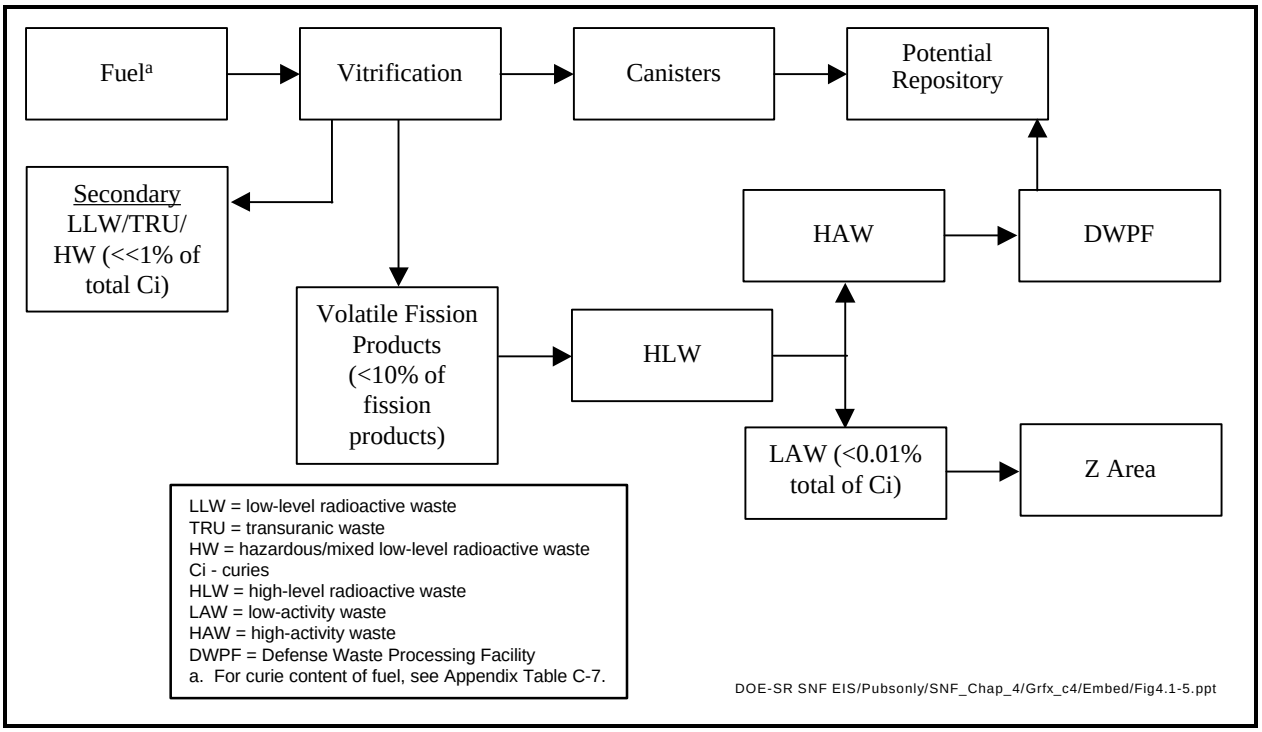

Figure 4.1-5. Type and source of waste streams generated by the Vitrification technology options.

EC

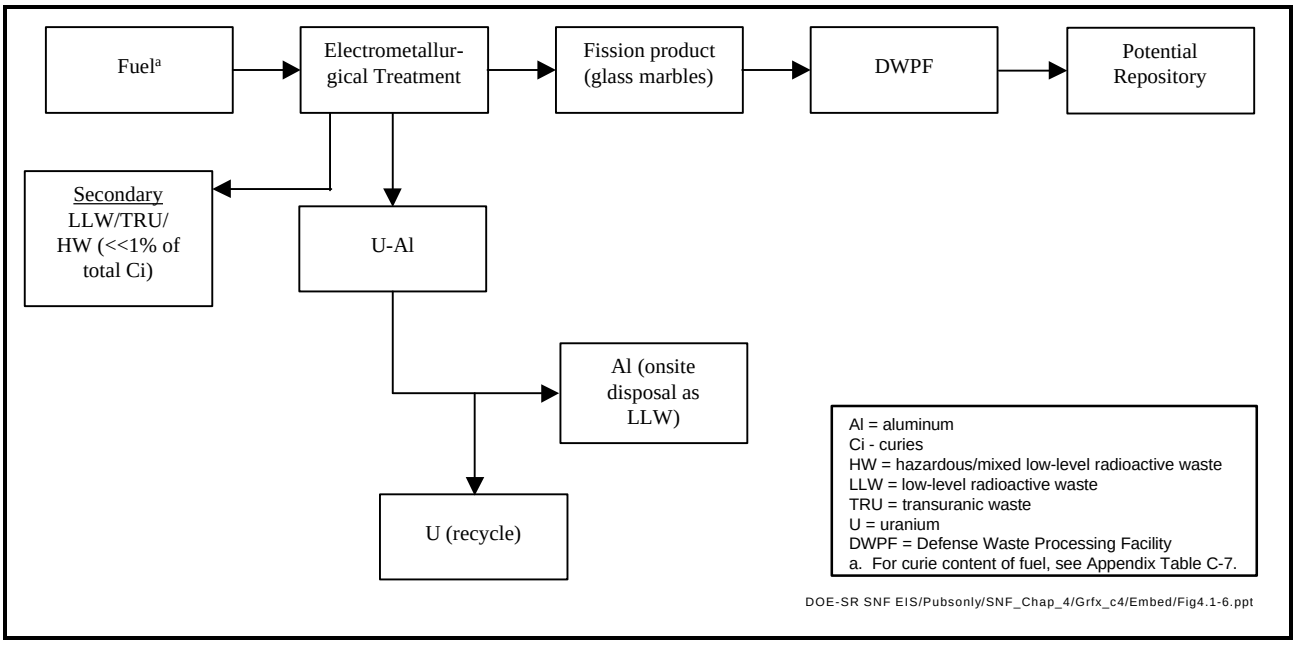

Figure 4.1-6. Type and source of waste streams generated by the Electrometallurgical Treatment technology option.

EC

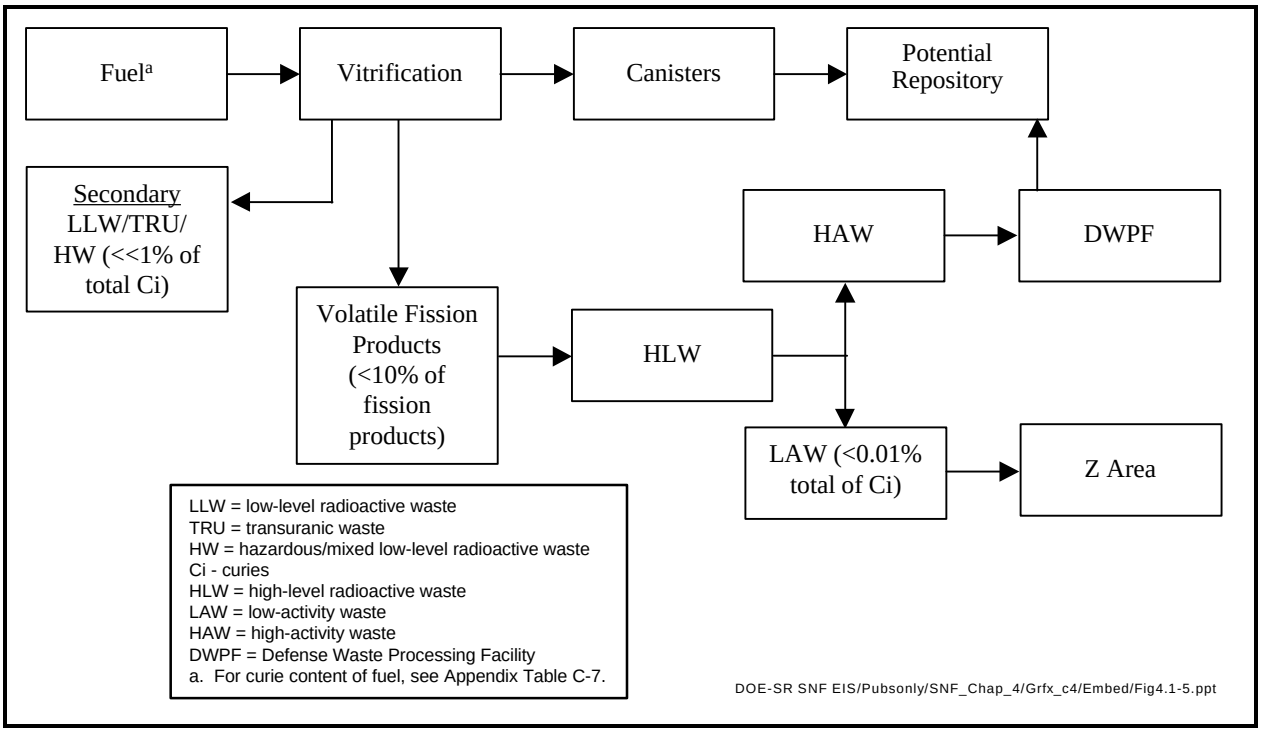

Figure 4.1-5. Type and source of waste streams generated by the Vitrification technology options.

EC

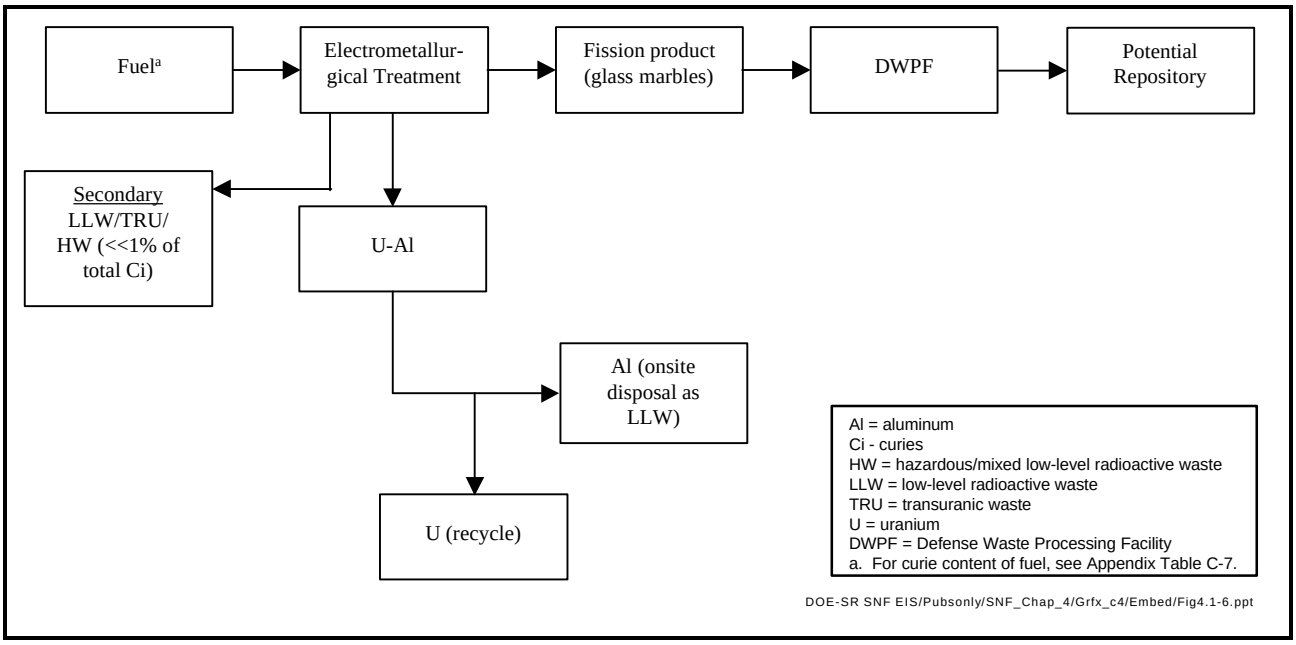

Figure 4.1-6. Type and source of waste streams generated by the Electrometallurgical Treatment technology option.

EC

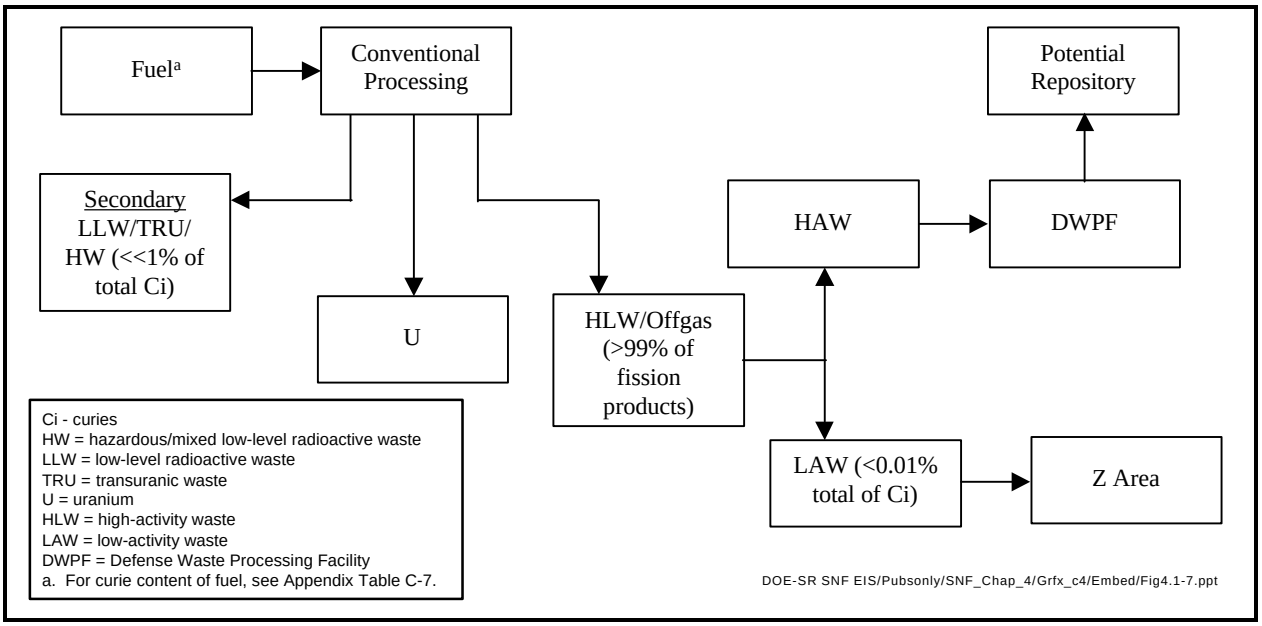

Figure 4.1-7. Type and source of waste streams generated by the Conventional Processing technology option.

rent and past SRS operations or on engineering judgments for new treatment technologies.

The following paragraphs describe estimated utility requirements for the options.

4.1.1.5.1 Water Use

Vitrification and Electrometallurgical Treatment would require the most water, followed by Conventional Processing. Total requirements for Vitrification and Electrometallurgical Treatment of all applicable fuel groups would be less than 6,000 liters per year, (the equivalent of 4.3 gallons per day) which is a minute portion (0.00001 percent) of groundwater withdrawal of more than 5×10^9 liters per year (DOE 1997). Due to the comparatively long period required to process the HEU/LEU oxides and silicides requiring resizing or special packaging (Fuel Group C) and the loose uranium oxide in cans (Fuel Group D), the Conventional Processing technology would require the greatest amount of water for those groups. For the higher actinide targets, Repackage and Prepare to Ship would require 67 percent of the water needed to support the only other option under consideration for that fuel group, Continued Wet Storage. In general, the Direct Disposal/Direct Co-Disposal, Melt and Dilute, Mechanical Dilution, and Repackage and Prepare to Ship technologies would require the least water for their applicable fuel groups, approximately 5 to 6 percent of the maximum requirement for a given group.

4.1.1.5.2 Electricity Use

Vitrification and Electrometallurgical Treatment would have the highest annual demand for electricity, followed by Conventional Processing. Differences in the time necessary to treat a fuel group under different options would affect total electricity requirements. Due to the longer period required to process the materials test reactor-like fuels (Fuel Group B), HEU/LEU oxides and silicides requiring resizing or special packaging (Fuel Group C), and loose uranium oxide in cans (Fuel Group D), Conventional Processing would require the most total electricity for those groups. For the higher actinide targets, Repackage and Prepare to Ship would require less than half the electricity needed to support continued wet storage. In general, for the appropriate fuel groups, the least electricity would be required to support Direct Co-Disposal and Mechanical Dilution.

CHAPTER 4. ENVIRONMENTAL IMPACTS

This chapter describes potential environmental impacts from construction, operation, and accidents associated with the proposed action and its alternatives. Section 4.1 describes the operational impacts of each alternative within the scope of this environmental impact statement (EIS). Section 4.2 describes risks to members of the public and onsite workers from potential facility accidents associated with the management of spent nuclear fuel (SNF) at the Savannah River Site (SRS). Section 4.3 describes impacts that could result from construction activities associated with SNF management at SRS. The purpose of the information presented in this chapter is to provide comparisons among alternatives. For new facilities, this information is based on DOE's best estimates of these facilities' operational characteristics. These data are not intended to be used for safety analysis purposes or compared to safety documents such as a Safety Analysis Report.

As discussed in Section 2.3.2, the Department of Energy (DOE) has identified three candidate sites for the potential construction of a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility: (1) the east side of L Area inside the facility fence, (2) the southeast side of C Area inside the facility fence, and (3) the northeast side of P Area. In addition, the facility could be constructed on a site inside the F-Area or H-Area fence or in an existing reactor building such as Building 105-L.

In most instances, implementing the technology options described in Chapter 2 would result in the same or very similar environmental impacts, regardless of location. If, during the preparation of this EIS, analyses indicated that a technology option would produce different environmental impacts at one of the candidate sites, DOE analyzed the site that would have the greatest impact (the bounding site). The analysis of the atmospheric releases of radioactivity described in the air resources and public and worker health sections is based on the assumption that emissions from a Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would occur in C Area. Releases from C Area would result in higher estimated radiation doses to members of the public than releases from L or P Area (i.e., C Area would result in doses to the maximally exposed offsite individual approximately 1.7 times higher than those in L Area and 1.1 times higher than those in P Area). All other impacts would be independent of location.

The impacts reported in this chapter are based on the entire SNF inventory described in Chapter 1 and Appendix C. However, as noted in Section 1.3, some foreign reactor operators may not participate in DOE's program of accepting U.S.-origin SNF. This reduction in receipts could potentially impact the amounts of fuel in Groups B, D, and E. Therefore, the amounts of fuel to be managed in those fuel groups could be less than the amounts assumed for the calculations in Chapter 4. DOE believes that annual impacts for normal operations, construction impacts, and accident impacts would be unaffected by modest reductions in the expected fuel inventory. The annual impacts are based on the maximum year's impacts; decreasing the foreign fuel shipments may lessen the number of years of fuel handling, conditioning, or treatment, but would not affect the maximum annual impact. SNF accidents usually involve small amounts of fuel and thus are insensitive to the total inventory. Construction impacts are similarly insensitive to the reduction in total fuel inventory that could occur. Eleven environmental impact measures are based on activities that occur over the entire period of analysis. These impacts would be sensitive to reductions in fuel receipts. Where applicable, the tables in this chapter explain how to adjust reported impacts for potentially reduced fuel receipts.

4.1 Impacts from Normal Operations

This section describes environmental impacts that could result from operational activities associated with SNF management at SRS for existing and new facilities. Because the only potential impacts to geologic and cultural resources would occur during construction (see Section 4.3), Section 4.1 does not consider geologic or cultural resource impacts. DOE does not anticipate a significant increase in employment due to the implementation of any technology options (Table 4.1-1). The existing site work force should be sufficient to provide the necessary operations and support personnel; therefore, there would be no socioeconomic impacts from operations under any technology.

Table 4.1-1. Estimated operational staffing for any of the technology options.

Technology option	Operations personnel	Support personnel	Total personnel
Melt and Dilute	200	200	400
Mechanical Dilution	175	175	350
Repackage and Prepare to Ship	75	75	150
Vitrification	317	317	634
Electrometallurgical	238	238	476
Conventional Processing	300	300	600
Continued Wet Storage	80	80	160

Source: Bickford et al. 1997.

DOE used the following process to estimate the impacts associated with new facilities/processes. First, DOE identified the facilities that would be needed to implement each of the technologies described in Chapter 2 (see Table 2-4). Next, DOE identified the major systems required within each facility for each technology. DOE then identified the energy sources, potential waste and effluent streams, and sources of potential radiation exposure associated with each of these major systems. These results were then compared to similar processes with which DOE has operational experience to determine the relative magnitude of the impact. These impacts were

presented as annual impacts; integrated impacts were then calculated as described below in Section 4.1.1.

DOE does not expect normal operations to have any appreciable impacts on ecological resources. Impacts would be limited to minor disturbances of animals in undeveloped areas adjacent to SNF management facilities caused by increased movement and noise from personnel, vehicles, and equipment. However, these impacts would be negligible under all proposed technology options because they would occur in areas where industrial activities already exist. Impacts to potential human receptors from normal releases of radioactive and nonradioactive contaminants to the environment would be small for any of the technologies under consideration (Section 4.1.1.3). Therefore, these releases would not be likely to produce measurable effects on nearby plant and animal communities or to accumulate in aquatic or terrestrial ecosystems.

4.1.1 IMPACTS OF TECHNOLOGY OPTIONS

This section describes the environmental impacts of each technology. The analysis covers the environmental impacts of actions over the 38-year period from 1998 through 2035 and presents both maximum annual impacts from these technologies and estimated total impacts over the entire period. For example, the discussions of water and air resources present maximum annual radiation doses to members of the public from liquid and airborne emissions associated with each technology and compares the resulting values to Federal limits. The section on public and worker health, on the other hand, presents radiation doses to members of the public from liquid and airborne emissions over the entire implementation period. The waste generation and utilities and energy sections also present impacts over the entire period of analysis (1998-2035). EC

To estimate total impacts, DOE identified the activities necessary to implement each technology, the amount of time required for each step (*phase*) of the technology option, and the annual impacts likely to occur during each phase. DOE summed the annual impacts over the entire duration of the phase, together with other phases needed to implement that option. For the Conventional Processing option, DOE used historic data for F- and H-Canyon operations to estimate the time needed to process the entire inventory of each type of fuel (McWhorter 1997). For the other technology options with a treatment phase, DOE used engineering judgments to estimate the duration of this phase for each fuel group. Appendix E describes the assumed durations for each phase. If annual impact data (i.e., utilities and energy, waste generation, and worker radiation dose) for each type of fuel were not available, DOE assumed that the fraction of the impact attributable to each type of fuel would be equal to the fraction of that fuel's fissile mass to the total fissile mass of SNF in the scope of this EIS. DOE derived the annual impact calculations from the available data (Bickford et al. 1997) based on the total radionuclide inventory for each type of fuel. Appendix C contains the radionuclide inventories, using a "reference fuel assembly" i.e., a conservative estimate of the radionuclide and curie content for an SNF assembly designed to bound the characteristics of fuel assigned to SRS. The engineering report that provides data upon which the impacts presented in this chapter are based (Bickford et al. 1997) is available for review at the DOE public reading room in Aiken, South Carolina.

4.1.1.1 Water Resources

This section describes the effects of normal operations associated with the technologies to SRS waters. All process water would come from groundwater. None of the technologies require much water to process the fuels. At most, less than 6,000 liters per year (equivalent to 1,585 gallons per year) would be required. The SRS annually withdraws more than 5×10^9 liters of groundwater (DOE 1997).

As discussed below, the only technology that would result in discharges of radionuclides or nonradioactive hazardous materials to surface water would be conventional processing. The

major sources of liquid effluents from facilities associated with conventional processing would be process cooling water and steam condensate that could contain small quantities of radionuclides and chemicals. Conventional processing would use wastewater treatment facilities and other equipment designed for full production (i.e., five production reactors, two separation facilities, and other industrial facilities) loads. Therefore, capacities would be sufficient to handle the liquid effluents and other secondary waste associated with conventional processing.

Liquid effluents associated with the SNF technologies would use existing wastewater treatment facilities and outfalls described in Section 3.2.1.3. Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility (CSWTF) and discharged through an existing NPDES outfall (G-10). Because technology options would not increase the number of permanent SRS employees, the CSWTF treatment rates would not be affected, and it would continue to meet the requirements of the SRS NPDES permit.

DOE evaluated in the Programmatic SNF EIS (DOE 1995b) the potential impacts to groundwater from a direct leak to the subsurface from a breach in a storage pool during routine operations. Because basin water could contain some radionuclides but would not contain any toxic or harmful chemicals, the following evaluation addresses only the consequences of radionuclide releases. The analysis conservatively assumed a 5-gallon (19-liter) per-day leak as a result of secondary containment or piping failure at the Receiving Basin for Offsite Fuels, L-Reactor Disassembly Basin, or a new wet receipt basin in a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility. The analysis assumed further that the leak would go undetected for 1 month.

The reliability and sensitivity of the leak detection devices at a new wet receipt basin would be equal to or superior to those required by the U.S. Nuclear Regulatory Commission (NRC 1975) for SNF storage facilities in commercial nuclear power plants. Constant process monitoring, mass balance, and facility design (including double-walled containment of vessels and piping) also would be used by DOE to limit operational releases from a new wet receipt facility to near zero.

A leak from the Receiving Basin for Offsite Fuels, or the L-Reactor Disassembly Basin, could result in the introduction of radionuclide-contaminated water into the ground at depths as much as 44 feet (13.4 meters) below grade. Such a release would go directly to the uppermost aquifer (Upper Three Runs), which at SRS is not suitable for use as a drinking water source because of its low yield and the presence of contaminants. Any contaminants would move through the Upper Three Runs and Gordon aquifers and ultimately discharge to SRS streams. The processes governing the plume movement (i.e., the hydraulic conductivity, hydraulic gradient, and effective porosity of aquifers in F, H, and the Reactor 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) would mitigate impacts to surface- or groundwater resources. Localized contamination of groundwater in the surface aquifer could occur in the immediate vicinity of the storage facility. However, this aquifer is not used as a source of drinking water. DOE concludes that no radionuclide contamination of deeper confined aquifers that are sources of onsite or offsite drinking water would be likely to occur from a leak in a storage basin.

The aquifer used as the primary source for drinking water is separated from the shallower aquifers by a confining unit. The hydraulic pressure of the lower aquifer is greater than that of the overlying aquifer. Therefore, water flows from the lower to the upper aquifer. This upward flow would prevent the downward migration of released contaminants.

4.1.1.1.1 Radiological Impacts

With the exception of conventional processing which is the maximum impact alternative, none of the technologies proposed in this EIS is likely to result in measurable increases in radionuclides released to water (Bickford et al. 1997). No other proposed technology would have a process discharge to surface waters.

The prolonged storage of SNF in the basins (i.e., the No-Action Alternative) could lead to a higher rate of fuel failures and releases to basin water, but probably would not affect routine releases (i.e., those from national pollutant discharge elimination system [NPDES] permitted outfalls). DOE would maintain water quality by monitoring basin water, deionizing basin water using resin beds, and stabilizing leaking assemblies.

Calculations of radiological doses through water pathways based on these releases are supported by the use of LADTAPXL, a spreadsheet version of the LADTAP II computer code developed by the U.S. Nuclear Regulatory Commission (NRC) to estimate radiation doses associated with normal reactor system liquid effluent releases to individuals, populations, and biota (Hamby 1991). LADTAP II uses the models in NRC Regulatory Guide 1.109 (NRC 1977) to calculate doses received from water and fish ingestion and from recreational water activities. Parameters used to calculate dose for the maximally exposed individual are consistent with regularly published SRS environmental reports (e.g., Arnett and Mamatey 1996).

Any radionuclide releases to surface water resulting from the technologies would be to SRS streams that discharge to the Savannah River. For all technology options, the ingestion of fish contaminated with cesium-137 would contribute most of the exposure to both the maximally exposed individual and the population. Plutonium and uranium isotopes ingested with drinking water would be smaller contributors for the approximately 70,000 people served by water treatment plants near Port Wentworth, Georgia (60,000) and Beaufort, South Carolina (10,000) (Arnett and Mamatey 1996). Table 4.1-2 lists both the maximally exposed individual dose and the collective dose due to liquid releases to the 620,100-person population surrounding SRS. TC

Table 4.1-2. Estimated maximum incremental annual dose to hypothetical maximally exposed individual and 620,100-person population surrounding SRS due to liquid releases from Conventional Processing.

Fuel group	MEI dose (millirem)	Population dose (person-rem)
A. Uranium and Thorium Metal Fuels	4.2×10^{-5}	2.4×10^{-4}
B. Materials Test Reactor-Like Fuels	0.042	0.14
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0.014	0.047
D. Loose Uranium Oxide in Cans	1.4×10^{-3}	4.7×10^{-3}
E. Higher Actinide Targets	NA	NA
F. Non-Aluminum-Clad Fuels	NA	NA

NA = Technology is not applicable to this fuel type.
HEU = Highly Enriched Uranium.
LEU = Low Enriched Uranium.
MEI = Maximally Exposed Individual.

4.1.1.1.2 Nonradiological Impacts

This assessment compared chemical releases with applicable water quality standards. These standards are based on the preservation of aquatic biota populations, human health, and aesthetics (i.e., taste and odor). Figure 3.2-1 shows that conventional processing activities would not occur in the 100-year floodplain. DOE would treat sanitary waste generated by any of the alternatives in this EIS in existing sewage treatment fa-

cilities; discharges from these facilities would continue to meet NPDES permit limits.

Activities associated with the New Packaging Technology options and all new treatment options under the New Processing Technology, including Melt and Dilute, Mechanical Dilution, Vitrification, and Electrometallurgical Treatment, would conform to current regulatory standards, and would not have nonradiological waterborne releases (Bickford et al. 1997). Under conventional processing, process cooling water treatment would result in releases of the following concentrations from F Area to Upper Three Runs:

- Nitrate - 40 micrograms per liter
- Ammonia - 30 micrograms per liter
- Manganese - 10 micrograms per liter
- Uranium - 20 micrograms per liter
- Nickel - 50 micrograms per liter
- Chromium - 20 micrograms per liter
- Aluminum - 200 micrograms per liter
- Copper - 10 micrograms per liter
- Zinc - 70 micrograms per liter

Similar or lower concentrations would be released from H Area with the exception of those for nitrate and ammonia, which would be 100 and 500 micrograms per liter, respectively.

Although proposed or final Federal drinking water standards do not apply to discharges, the SRS discharge concentrations would not exceed these standards. The discharges would also comply with South Carolina Water Quality Standards contained in South Carolina Regulation R.61-68. In general, the release concentrations would be no greater than those currently measured in Upper Three Runs and Fourmile Branch (Arnett 1996), with the exception of zinc and ammonia; however, zinc concentrations in the discharge would be only a small fraction of the South Carolina Water Quality Standards, which are based on the taste and odor of drinking water. Ammonia concentrations in the discharge (only H-Area releases would increase current stream concentrations) would be well within state standards. Lead, nickel, and chromium generally were not detected in Upper Three Runs and Fourmile Branch in 1995.

4.1.1.2 Air Resources

This section describes incremental air quality impacts from nonradiological and radiological emissions for the operation of each technology option for each fuel group; this description includes impacts to on- and offsite individuals and populations.

This analysis presents results in terms of ground-level air concentrations for nonradiological constituents and radiation dose for radionuclides because these are the best measures of potential adverse human health effects.

4.1.1.2.1 Nonradiological Emissions

DOE estimated nonradiological emission rates for each technology option (Bickford et al. 1997) and used them with the meteorological data described in Section 3.3.1 to estimate site boundary and noninvolved worker concentrations. This analysis assumed average meteorological conditions.

Onsite Concentrations

The purpose of this analysis is to estimate air concentrations to which SRS workers not involved in SNF management and related operations would be exposed. Atmospheric emissions would occur from F or H Area (conventional processing), L-Reactor Disassembly Basin and the Receiving Basin for Offsite Fuels (continued wet storage), and the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility. To determine impacts to noninvolved workers, the analysis used a generic location 2,100 feet (640 meters) from the release in the direction of the plume of greatest concentration. The 2,100-foot criterion is based on NRC guidance. Also, the use of this distance ensures consistency between this and previous SRS EISs.

The analysis assumed that operational nonradiological releases would be from the same release stack as radiological releases. In addition, this EIS does not include onsite concentrations at distances greater than 2,100 feet; the analysis considered such concentrations and found that they would be less than those at 2,100 feet.

Tables F-1 through F-10 in Appendix F list estimated air concentrations above baseline (i.e., incremental increases) resulting from nonradiological atmospheric emissions associated with SNF fuel groups. No incremental atmospheric emissions above the baseline presented in Chapter 3 would be associated with Repackage and Prepare to Ship, the only option applicable to the non-aluminum-clad fuels. The air quality regulatory standards listed in Tables F-6 through F-10 in Appendix F are applicable to the Site boundary concentration from all SRS emissions. While these standards are included only for reference, all the incremental concentrations from SNF activities would be at least two orders of magnitude less than any of the corresponding standards except those for nitric acid, oxides of nitrogen, and gaseous fluorides emitted during conventional processing or vitrification of fuel Group B. The concentrations would range from less than 1 percent to about 55 percent of the offsite standard (for nitrogen oxides). If a new facility or a major modification to an existing facility were being considered, new permitting actions would be required as part of the Clean Air Act Title V permit compliance requirements. Under the current Title V permit, SRS would have to conduct a Prevention of Significant Deterioration review, since the nitrogen oxide levels exceed the 25 μm per cubic meter per year threshold of NO_2 for a Class II area. In addition, there would be a requirement for ambient monitoring to verify emission levels once the process began.

Offsite Concentrations

This analysis presents projected maximum offsite nonradiological incremental air concentrations in much the same way it presents the onsite concentrations. The estimated maximum incremental concentrations listed in Tables F-6 through F-10 in Appendix F would occur at the SRS boundary for emissions associated with SNF. The air quality regulatory standards listed in the tables are applicable to the Site boundary concentrations from all SRS emissions. All the incremental concentrations are at least three orders of magnitude less than any of the corresponding standards except those for oxides of nitrogen and gaseous fluorides emitted during conventional processing or vitrification. The concentrations ranged from less than 1 percent to about 2 percent of the offsite standard.

4.1.1.2.2 Radiological Emissions

DOE estimated airborne radionuclide emission rates for each technology option (Bickford et al. 1997), and used them with the meteorology data from Section 3.3.1 as inputs to the SRS com-

puter models MAXIGASP and POPGASP (Hamby 1994) to determine doses to onsite (noninvolved worker) and offsite (hypothetical maximally exposed individual) recipients and the surrounding population (620,000 persons) within a 50-mile (80-kilometer) radius of the center of the Site (Simpkins 1996). The analysis uses the meteorological data to determine annual average concentrations in air. The values presented in Tables 4.1-3, 4.1-4, and 4.1-5 represent current reactor-area emissions (including two SNF wet basins).

Onsite Doses

Atmospheric doses to the noninvolved worker represent the radiological exposures of a hypothetical worker who is nearby but not involved in SNF operations. Table 4.1-3 lists the estimated maximum incremental annual doses to noninvolved workers from atmospheric emissions of radionuclides for each viable technology option for each fuel group. The EPA limit of 10 millirem per year (40 CFR Part 61, Subpart H) is a point of comparison for these doses. (In fact, this limit is applicable to offsite individuals from sitewide airborne releases; see Chapter 5). The highest incremental dose to the noninvolved worker would be 0.27 millirem (from Melt and Dilute, Vitrification, or Electrometallurgical Treatment of Materials Test Reactor-like Fuels). Incremental doses to the noninvolved worker from all viable options would be 3 percent or less of the national emission standards for hazardous air pollutants (NESHAP) limit.

EC

There would be no pathways for exposure of personnel inside SNF management facilities from atmospheric releases of radioactivity. Section 4.1.1.3 discusses radiation doses to SNF management workers, including from in-facility airborne releases of radioactivity.

Offsite Doses

Atmospheric doses to the hypothetical maximally exposed offsite individual assume a person who resides at the SRS boundary at the point of maximum exposure. Every member of the public would have a dose less than that received by this individual. Table 4.1-4 lists the estimated maximum incremental annual dose to this individual from atmospheric emissions of radionuclides for each technology option for each fuel group. As with the doses to noninvolved workers, the NESHAP limit of 10 millirem per year (40 CFR Part 61, Subpart H) is a point of comparison. The maximum incremental annual dose from any technology option for a given fuel group would be 0.033 millirem per year (from Melt and Dilute, Vitrification, or Electrometallurgical Treatment of Materials Test Reactor-like Fuels), a factor of 300 less than the EPA limit.

Table 4.1-5 lists the estimated maximum incremental annual population dose (the collective dose to the entire population around SRS) for each viable option. The maximum incremental annual population dose from any option would be 1.2 person-rem per year (from Melt and Dilute, Vitrification, or Electrometallurgical Treatment of Materials Test Reactor-like Fuels).

4.1.1.3 Worker and Public Health

This section discusses potential radiological and nonradiological health effects to SRS workers and the surrounding public from the technology options for the management of SNF; it does not include impacts of potential accidents, which are discussed in Section 4.2. DOE based its calculations of health effects from the air- and waterborne radiological releases on (1) the dose to the hypothetical maximally exposed individual

in the public; (2) the collective dose to the population within a 50-mile (80-kilometer) radius around the SRS (approximately 620,000 people); (3) the collective dose to workers involved in implementing a given alternative (i.e., the workers involved in SNF management activities); and (4) the dose to the maximally exposed noninvolved worker (i.e., SRS employees who may work in the vicinity of the SNF management facilities but are not directly involved in SNF work). All radiation doses mentioned in this EIS are effective dose equivalents; internal exposures are committed effective dose equivalents. This section presents total impacts for the entire length of time necessary to implement each technology, using the durations listed in Appendix E. The annual impacts attributable to each phase were multiplied by the duration of that phase. The impacts from all phases were summed to calculate the total impact for the technology. This discussion characterizes health effects as additional lifetime latent cancer fatalities likely to occur in the general population around SRS and in the population of workers who would be associated with the options.

4.1.1.3.1 Radiological Health Effects

Radiation can cause a variety of health effects in people. The major effects that environmental and occupational radiation exposures could cause are delayed cancer fatalities, which are called latent cancer fatalities because the cancer can take many years to develop and cause death.

To relate a dose to its effect, DOE has adopted a dose-to-risk conversion factor of 0.0004 latent cancer fatality per person-rem for workers and 0.0005 latent cancer fatality per person-rem for the general population (NCRP 1993). The factor for the population is slightly higher due to the presence of infants and children who might be more sensitive to radiation than workers, who are, generally speaking, healthy adults.

DOE uses these conversion factors to estimate the effects of exposing a population to radiation. For example, in a population of 100,000 people exposed only to background radiation (0.3 rem per year), DOE would calculate 15 latent cancer fatalities per year caused by radiation (100,000 persons $\times$ 0.3 rem per year $\times$ 0.0005 latent cancer fatality per person-rem).

Calculations of the number of latent cancer fatalities associated with radiation exposure might not yield whole numbers and, especially in environmental applications, might yield values less than 1. For example, if a population of 100,000 were exposed only to a dose of 0.001 rem to each person, the collective dose would be 100 person-rem, and the corresponding number of latent cancer fatalities would be 0.05 (100,000 persons $\times$ 0.001 rem $\times$ 0.0005 latent cancer fatality per person-rem).

DOE also has employed these concepts in estimating the effects of radiation exposure to a single individual. For example, consider the effects of exposure to background radiation over a lifetime. The number of latent cancer fatalities corresponding to an individual's exposure over a (presumed) 72-year lifetime at 0.3 rem per year would be 0.011 latent cancer fatality (1 person $\times$ 0.3 rem per year $\times$ 72 years $\times$ 0.0005 latent cancer fatality per person-rem).

This number should be interpreted in a statistical sense; that is, the estimated effect of background radiation exposure to the exposed individual is a 1.1-percent lifetime chance that the individual might incur a latent fatal cancer. Vital statistics on mortality rates for 1994 (CDC 1996) indicate that the overall lifetime fatality rate in the United States from all forms of cancer is about 23.4 percent (23,400 fatal cancers per 100,000 deaths).

These factors, which DOE uses in this EIS to relate radiation exposure to latent cancer fatalities, are based on the *Recommendations of the International Commission on Radiation Protection* (ICRP 1991). They are consistent with the factors used by the U.S. Nuclear Regulatory Commission in its rulemaking *Standards for Protection Against Radiation* (10 CFR Part 20). The factors apply if the dose to an individual is less than 20 rem and the dose rate is less than 10

rem per hour. At doses greater than 20 rem, the factors used to relate radiation doses to latent cancer fatalities are doubled. At much higher dose rates, prompt effects, rather than latent cancer fatalities, would be the primary concern.

In addition to latent cancer fatalities, other health effects could result from environmental and occupational exposures to radiation; these include nonfatal cancers among the exposed population and genetic effects in subsequent generations. Previous studies have concluded that these effects are less probable than fatal cancers as consequences of radiation exposure (ICRP 1991). Dose-to-risk conversion factors for nonfatal cancers and hereditary genetic effects (0.0001 per person-rem and 0.00013 per person-rem, respectively) are substantially lower than those for fatal cancers. This EIS presents estimated effects of radiation only in terms of latent cancer fatalities because that is the major potential health effect from exposure to radiation. Estimates of nonfatal cancers and hereditary genetic effects can be estimated by multiplying the radiation doses by the effects dose-to-risk conversion factors.

DOE expects minimal worker and public health impacts from the radiological consequences of managing SNF under any of the technology options, as well as Continued Wet Storage. However, some options would result in increased radiological releases. Public radiation doses include doses from airborne releases (Section 4.1.1.2) and liquid releases (Section 4.1.1.1). Table 4.1-6 lists incremental radiation doses estimated for the public (maximally exposed individual and collective population dose) and corresponding incremental latent cancer fatalities, for each fuel group and technology option.

The values in Tables 4.1-6 and 4.1-8 for the No-Action Alternative represent current reactor-area emissions (including two SNF wet basins) for the entire period of analysis. The values for the other alternatives would be incremental above these baseline values. Summing these baseline and incremental values would be conservative, however, because there would not be two SNF wet basins operating over the entire 38-year period of analysis.

DOE based estimated worker doses on past operating experience and the projected durations for implementation of the alternative actions (Bickford et al. 1997). For the maximally exposed worker, DOE assumed that no worker would receive an annual dose greater than 500 millirem from any option because SRS uses the 500-millirem value as an administrative limit for normal operations; that is, an employee who receives an annual dose approaching the administrative limit normally is reassigned to duties in a nonradiation area. (Note: If DOE privatized the Transfer and Storage Facility or treatment operations, the licensee would adopt NRC worker dose limits, and administrative limits could be subject to adjustment.) Tables 4.1-7 and 4.1-8 estimate radiation doses for the collective population of workers who would be directly involved in implementing the options and for the noninvolved worker (a worker not directly involved with implementing the option but located 2,100 feet [640 meters] from the SNF facility) for each fuel group and technology option. These tables also list the latent cancer fatalities likely attributable to the doses. TC

Of the fuels considered for treatment (all except higher actinide targets and non-aluminum clad fuel), the highest expected radiological health effects to the public generally would occur under conventional processing. The single exception would be fewer latent cancer fatalities predicted for the population from the conventional processing of uranium and thorium metal fuels (Table 4.1-6). For the noninvolved workers, the conventional processing of Groups C and D fuels would result in the greatest radiological health effects. No measurable incremental increases would be likely for the higher actinide targets or the non-aluminum-clad fuels for any option because the only options applied to those groups are repackaging and continued wet storage. The estimated collective dose for workers who would be directly involved in managing SNF (Table 4.1-7) depends largely on the difference in the number of workers involved in each option and

not on the difference in the amount of radioactivity.

The estimated number of latent cancer fatalities in the public listed in Table 4.1-6 can be compared to the projected number of fatal cancers (145,100) in the public around the SRS from all causes (as discussed in Section 3.7.1). Similarly, the estimated number of latent cancer fatalities in the worker population can be compared to the number in the worker population from all causes (approximately 23.4 per- cent; see Section 3.7.1). In all cases, the incremental impacts from the options would be negligible.

4.1.1.3.2 Nonradiological Health Effects

DOE evaluated the range of chemicals to which the public and workers would be exposed due to SNF management activities and expects minimal health impacts from nonradiological exposures. Section 4.1.1.1.1 discusses offsite chemical concentrations from air emissions. DOE estimated worker impacts and compared them to Occupational Safety and Health Administration (OSHA) permissible exposure limits (PELs) or ceiling limits for protecting worker health, and concluded that all impacts would be well below the limits.

OSHA limits (29 CFR Part 1910.1000) are time-weighted average concentrations that a facility cannot exceed during a prescribed duration of a 40-hour week. The facility cannot exceed OSHA ceiling concentrations during any part of the workday. These exposure limits refer to airborne concentrations of substances and represent conditions under which nearly all workers could be exposed day after day without adverse health effects. However, because of the wide variation in individual susceptibility, a small percentage of workers could experience discomfort from some substances at concentrations at or below the permissible limit. Table 4.1-9 summarizes the values of Permissible Exposure Limits that DOE compared to the data in Tables F-1 through F-5 in Appendix F.

4.1.1.4 Waste Generation

This section presents waste generation estimates for each technology option and fuel group that DOE considers in this EIS. Tables 4.1-10 through 4.1-13 list these estimates. For each technology option, this analysis considered three handling phases as potential sources of waste: wet storage (pretreatment storage), treatment or conditioning, and dry storage (post-treatment storage pending final disposition). The period and waste generation rate associated with each phase varied depending on the fuel group and the technology. As discussed above, DOE summed waste volumes from each phase; the values listed in the tables represent the total projected waste volumes for each technology option in a given fuel group.

DOE used the annual waste generation rates to calculate the estimates in the tables (Bickford et al. 1997); the rates are based on applicable current and past SRS operations or on process

Table 4.1-9. Permissible Exposure Limits (milligrams per cubic meter) of nonradiological air pollutants regulated by the Occupational Safety and Health Administration.[a]

Pollutant	Averaging time	OSHA PEL[b]
Carbon monoxide	8 hours	55
Nitrogen oxides	1 hour	9[c]
Sulfur dioxide	8 hours	13
Carbon dioxide	8 hours	9,000
Nitric acid	8 hours	5

a. Source: 29 CFR Part 1910.1000.
b. Occupational Safety and Health Administration (OSHA) permissible exposure limit (PEL).

c. OSHA ceiling limit not to be exceeded at any time during the workday.

knowledge for new treatment technologies. The operating history that was the basis for these estimates would maximize projected waste generation rates. As described in Section 3.8, the Site generates several types of waste (high-level, transuranic, mixed, hazardous, low-level, and sanitary). Wastes generated by SNF management activities would be comparable to wastes the SRS currently handles and would, therefore, not require unique treatment, storage, or disposal actions. This section does not consider sanitary waste, the production of which would be in direct proportion to the number of employees, because none of the technologies would increase the number of permanent Site employees.

DOE has implemented an aggressive waste minimization and pollution prevention program at SRS at the sitewide level and for individual organizations and projects. As a result, significant reductions have been achieved in the amounts of wastes discharged into the environment and sent to landfills, resulting in significant cost savings.

To implement a waste minimization and pollution prevention program at the SNF management facilities, DOE would characterize waste streams and identify opportunities for reducing or eliminating them. Emphasis would be placed on minimizing the largest waste stream, low-level L1-5 waste, through source reduction and recycling. Selected waste minimization practices could include: (1) process design changes to reduce the potential for spills and to minimize contamination areas, (2) decontamination of equipment to facilitate reuse, (3) recycling metals and other usable materials, especially during the construction phase of the project, (4) preventive maintenance to extend process equipment life, (5) modular equipment designs to isolate potential failure elements to avoid changing out entire units, and (6) use of non-toxic or less toxic materials to prevent pollution and minimize hazardous and mixed waste streams.

EC The following sections describe the differences in waste generation by waste type among the SNF management technologies considered in this EIS.

4.1.1.4.1 High-Level Waste

SRS reports high-level waste as liquid high-level waste, and in the related quantities of equivalent Defense Waste Processing Facility (DWPF) canisters and saltstone. The volume estimates for liquid high-level waste reported in Table 4.1-10 are for volumes as they leave the process and enter the high-level waste tanks. While it is necessary to consider this volume when evaluating the interim storage of high-level waste in the tank farms, the volume of liquid high-level waste is not meaningful when considering the storage and disposition of final waste forms. The liquid waste is evaporated and concentrated in the high-level waste tanks. The generation of secondary waste in the high-level waste tanks and DWPF, including waste generated as a result of activities described in this SNF EIS, is evaluated in the DWPF Supplemental EIS (DOE 1994). Therefore, capacity for management of SNF secondary waste in the tank farms and DWPF is provided within the scope of DWPF operations. DWPF canisters and saltstone are the product of liquid high-level waste treatment and evaporation and would be the basis for final storage and disposition considerations. Because the production of saltstone and DWPF canisters from a given liquid waste volume are generally proportional, this discussion applies equally to DWPF canisters and saltstone. For Conventional Processing, DWPF canisters would be the only product to be disposed in a geologic repository. TC

Conventional Processing is the only option that would generate significant quantities of high-level waste during the treatment phase. Each option would produce high-level waste during the wet storage phase and technologies such as melt and dilute, that require off-gas collection systems, would also produce high-level waste, but the quantity produced generally would be much lower than that associated with Conventional Processing. The waste generated during wet storage and new technology processing operations would not meet the formal definition of high-level waste (waste resulting from the processing of SNF), but would consist of such items as deionizer backwash and off-gas collection

products, which the SRS typically manages (or would manage) as high-level waste. The lengthy period associated with continued wet storage generally would make it the second largest producer of high-level waste. For the higher actinide targets, Conventional Processing was not considered, making Continued Wet Storage the greatest potential for high-level waste production. The volumes of high-level waste generated by the other options would vary depending on the duration of storage and the amount of fissile material in the fuel, but would be fairly comparable within a given fuel type and substantially less than the volumes associated with conventional processing. In addition, the condition of the fuel would influence the high-level waste generation rate (i.e., fuel in poor condition would result in higher generation of deionizer backwash).

Based on the capacities of the high-level waste tank farms and the current volume of high-level waste in storage (see Table 3.8-2), these projected high-level waste volumes probably would not require additional treatment and storage facilities beyond those currently available at SRS. DOE bases this conclusion on continued removal and treatment of the existing tank farm inventory. DWPF would be available to treat these projected high-level waste volumes.

4.1.1.4.2 Transuranic Waste

For all applicable fuel types, conventional processing would produce the largest volume of transuranic waste due to a higher generation rate and a longer processing time. Conventional processing of all applicable fuel groups would generate 3660 cubic meters of transuranic waste which is 29 percent of the total SRS transuranic waste generation forecast (Table 3.8-1). The next largest quantity that could be generated would be from the Vitrification and Electrometallurgical Treatments of all applicable fuel groups. Those technologies would generate 700 cubic meters of transuranic waste over the life of the project, which is less than 6 percent of the total SRS transuranic waste generation forecast. These two technologies would produce 9 to 37 percent of that produced by conventional processing, depending on the fuel group.

None of the treatment options associated with the higher actinide targets or non-aluminum-clad fuels would produce transuranic waste.

4.1.1.4.3 Hazardous/Low-Level Mixed Waste

For this EIS analysis, DOE grouped hazardous and low-level mixed wastes together because none of the options is likely to produce significant quantities of either.

The highest hazardous/low-level mixed waste generation rates would be associated with Vitrification and Electrometallurgical Treatments, followed by Mechanical Dilution. However, due to the longer time required to process the loose uranium oxide in cans, the Materials Test Reactor-like fuels, and the highly enriched uranium/low enriched uranium (HEU/LEU) oxides and silicides requiring resizing or special packaging, conventional processing would produce the largest volume of hazardous or mixed waste for those fuel groups. Vitrification and Electrometallurgical Treatments generally would produce the next largest quantities (35 to 88 percent of that produced by conventional processing, depending on the fuel group). For the uranium and thorium metal fuels, Vitrification and Electrometallurgical Treatments produce the largest quantities of hazardous/low-level mixed waste, followed by conventional processing. For applicable fuel groups, the Direct Disposal/Direct Co-Disposal technology would consistently produce the smallest quantities of hazardous or mixed waste. The waste volumes that continued wet storage or the Melt and Dilute technology would produce would be roughly comparable and generally intermediate among the technologies. For the higher actinide targets, the two technologies being considered (Repackage and Prepare to Ship and Continued Wet Storage) would produce small, comparable quantities of hazardous or mixed waste.

When all applicable technologies are considered, conventional processing would generate

the largest volume (264 cubic meters) of hazardous and low-level mixed waste, which is less than 1 percent of the 30-year forecast.

4.1.1.4.4 Low-Level Waste

The Direct Disposal/Direct Co-Disposal and Repackage and Prepare to Ship technology options would produce the least low-level waste. The Mechanical Dilution and Melt and Dilute options would produce intermediate quantities of low-level waste, between 9 and 37 percent of the maximum volume generated and within approximately 150 percent of the minimum volume, depending on the fuel group. For applicable fuel groups, conventional processing would produce the most low-level waste. In each case, continued wet storage would produce the next highest volume due to the combined effect of storage time and generation rate. When all applicable fuel groups are included, conventional processing would generate 138,200 cubic meters of low-level waste (29 percent of the SRS low-level waste 30-year forecast) and continued wet storage would generate 56,650 cubic meters (12 percent of the forecast). Of the two options being considered for the higher actinide targets, the Repackage and Prepare to Ship option would produce the smallest quantity of low-level waste, 32 percent of that estimated for Continued Wet Storage.

4.1.1.4.5 By-products of converting SNF into a waste form that is suitable for disposal in a geologic repository

With the exception of continued wet storage under the No-Action Alternative, the technology options would convert the fuels into a waste form that is likely to be suitable for permanent disposal in a geologic repository. The radioactive inventory in the final waste form would be substantially greater than 99 percent of the original fuel inventory. Very small amounts of residual radioactivity would remain in secondary low-level, hazardous/mixed low-level, and transuranic waste streams as illustrated in Figures 4.1-1 through 4.1-7. SRS would use the surplus capacity in existing waste management facilities to treat, store, dispose of, or recycle the secondary waste in accordance with applicable regulations.

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The melt and dilute and vitrification technologies would release from the fuel matrix volatile fission products (primarily cesium) from the fuel matrix which would be recovered as illustrated in Figure 4.1-3 and Figure 4.1-5. Residual cesium, strontium, and plutonium from conventional processing (as well as volatile fission products from melt and dilute, and vitrification technology options) would be moved from the high-level waste tanks and separated into a high volume – low radioactivity salt stream and a low volume – high radioactivity slurry. The salt stream would be approximately 95 percent of the total (before separation) volume and the slurry would capture approximately 99.999 percent of the cesium, strontium, and plutonium activity (Choi 1992). The slurry would be encapsulated in glass and poured into canisters at the Defense Waste Processing Facility The canisters would be stored in a Glass Waste Storage Building for ultimate disposal in a geologic repository. The salt stream would be mixed into and solidified with concrete and disposed of in the Z-Area vaults.

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4.1.1.4.6 Spent Fuel Canisters

DOE does not consider the SNF canisters resulting from alternate technology options to constitute a waste stream because they would be the end product of the new packaging options or new processing technology options being proposed. Nevertheless, the number of canisters is a useful measure of onsite storage space needed and the volume of the material that, after processing, could possibly be placed in a repository. Table 4.1-14 indicates the numbers of two types of canisters for the various technologies. The 17-inch canister would be used for co-disposal. The 24-inch canister would be used when the technology produces a vitrified product identical

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Table 4.1-14. Numbers of spent fuel co-disposal and high-level waste canisters.

Technology	Co-Disposal or Direct Disposal canisters	24-inch high-level waste canisters
Prepare for direct co-disposal	1,400	NA[a]
Repackage and prepare to ship	NA[b]	1
Melt and dilute	400	10
Mechanical dilution[c]	630	10
Vitrification technologies[d]	1,350	10
Electrometallurgical treatment	–	90
Conventional processing[e]	–	150
Continued wet storage	–	41

a. NA = not applicable, since DOE would use Co-Disposal.
b. Canisters would not be required to transfer material to another site.
c. Values were calculated for the press and dilute technology.
d. Values represent dissolve and vitrify and glass material oxidation and dissolution system technologies. The plasma arc technology would produce 490 canisters.
e. Values are for conventional processing the entire SNF inventory.

to the DPWF high-level waste borosilicate glass. After conventional processing, the 24-inch canisters would be stored in DWPF's Glass Waste Storage Building. The number of high-level waste canisters (Table 4.1-14) includes the secondary waste stream components generated by the technologies reported in Table 4.1-10.

4.1.1.5 Utility and Energy Resources

This section describes the estimated utility and energy requirements associated with each technology option under consideration in this EIS. Water, electricity, steam, and diesel fuel would be required to support many of the options. Estimates of water use include domestic water supplies and makeup water for process operations or equipment cooling. Steam is used primarily to heat facilities. Fuel consumption is based on use of diesel generators for backup power. Electrical requirements include that for normal office consumption such as heating, cooling, ventilation, and office equipment, and for specialized process-related equipment. The process equipment and the associated electrical demands would vary from option to option. All technologies would require canister loading and welding equipment. For the Melt and Dilute technology, the resistive heating associated with melting would require additional electricity. For aqueous processing, electrical requirements would include the operation of canyon pumps, circulators or mixers, and denitriting equipment. For Vitrification, electrical equipment would be used for resistive heating and dissolution. For Electrometallurgical Treatment, electricity would be used for resistive melting of fuels, operation of an electrolytic bath for metal purification, final melting of the refined uranium product, and blending down with depleted uranium.

Tables 4.1-15 through 4.1-18 list estimated utility and energy requirements for the technology options applicable to each fuel group. For each option, this analysis considered three handling phases as potential sources of energy consumption: wet storage (pretreatment storage), treatment, and dry storage (post-treatment storage pending final disposition). The durations for these phases are provided in Appendix E. The period and utility use rate associated with each phase would vary depending on the fuel group and the option. As discussed above, DOE summed utility use from each phase; the values listed in the tables represent the total projected utility use for each option in a given fuel group.

DOE used annual utility consumption rates to calculate the estimates in the tables (Bickford et

al. 1997); the rates are based on applicable cur-

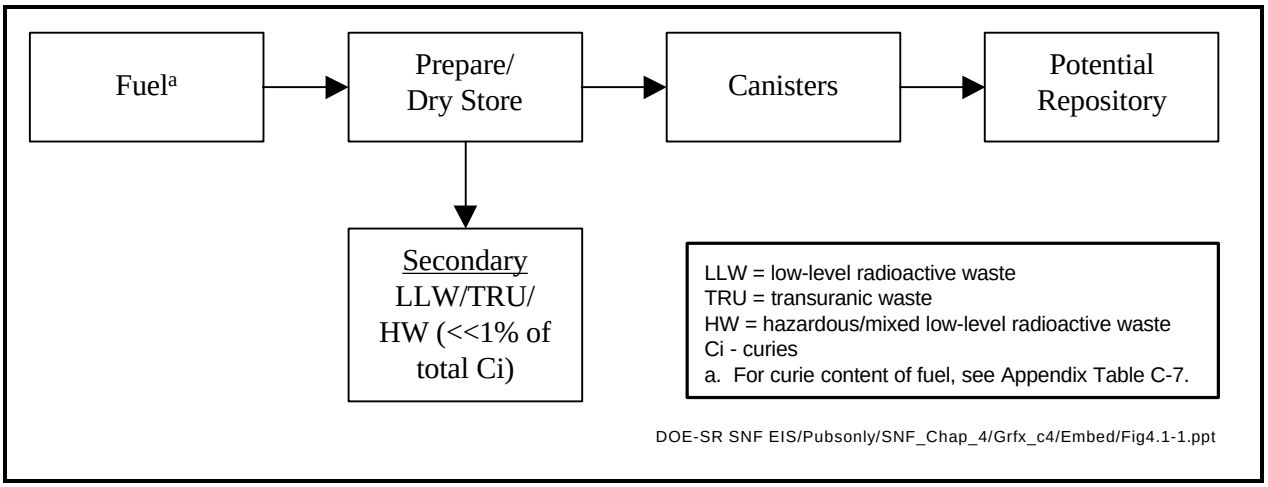

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Figure 4.1-1. Type and source of waste streams generated by the Prepare for Direct Co-Disposal technology option.

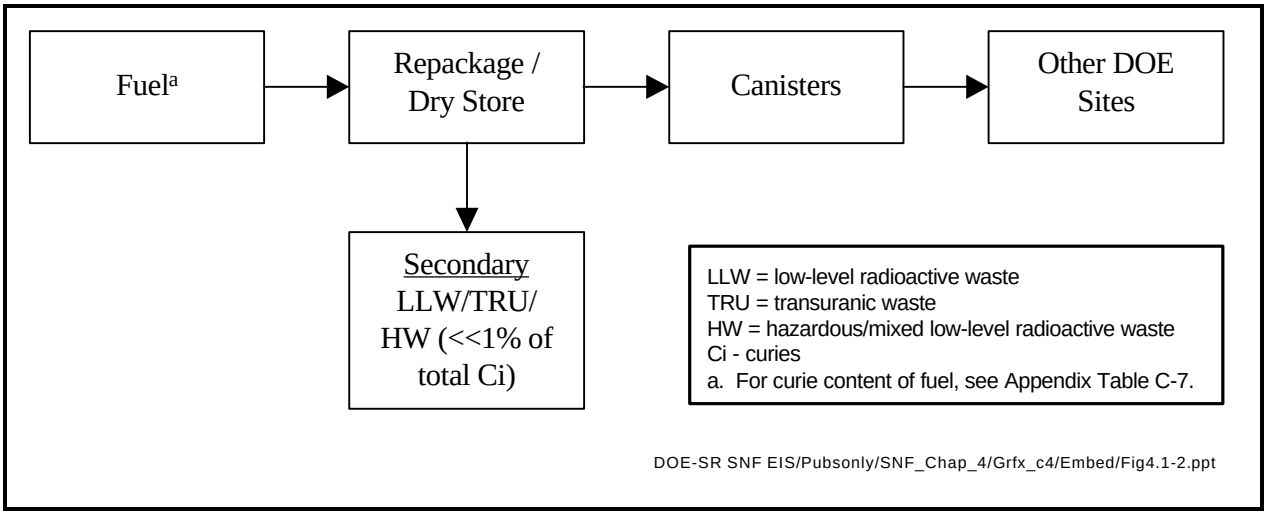

Figure 4.1-2. Type and source of waste streams generated by the Repackage and Prepare to Ship technology option.

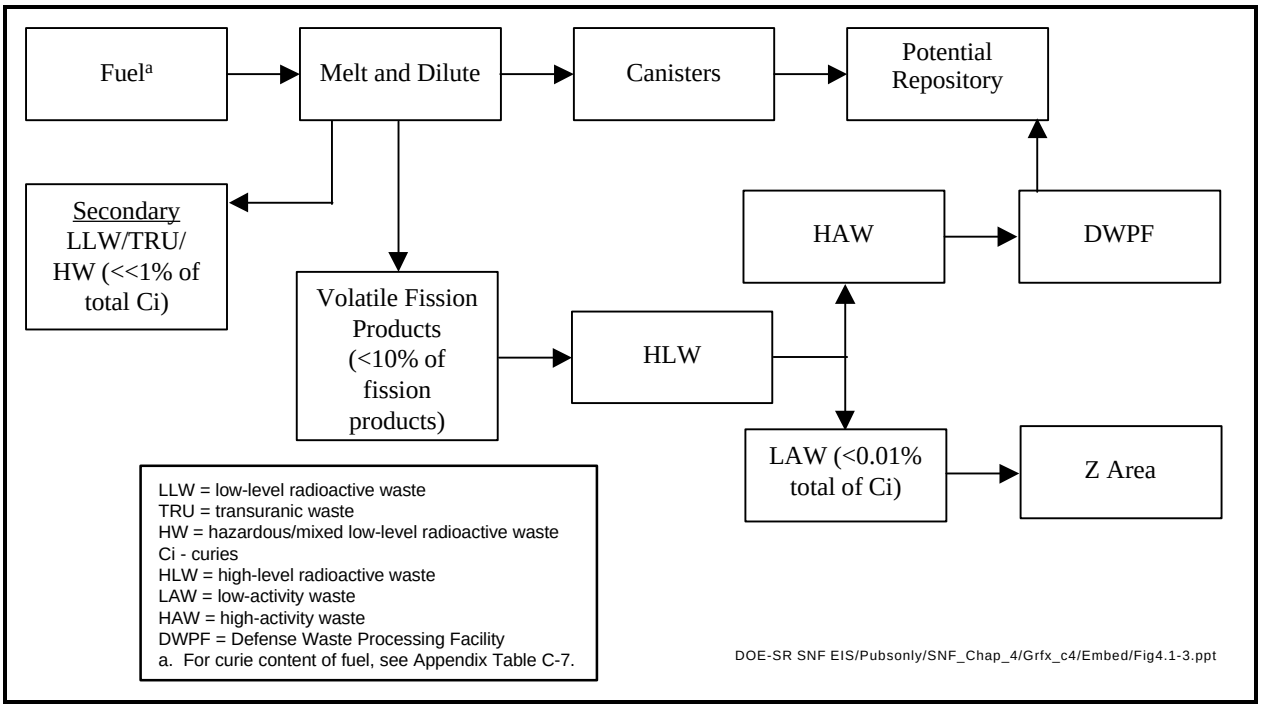

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Figure 4.1-3. Type and source of waste streams generated by the Melt and Dilute technology option.

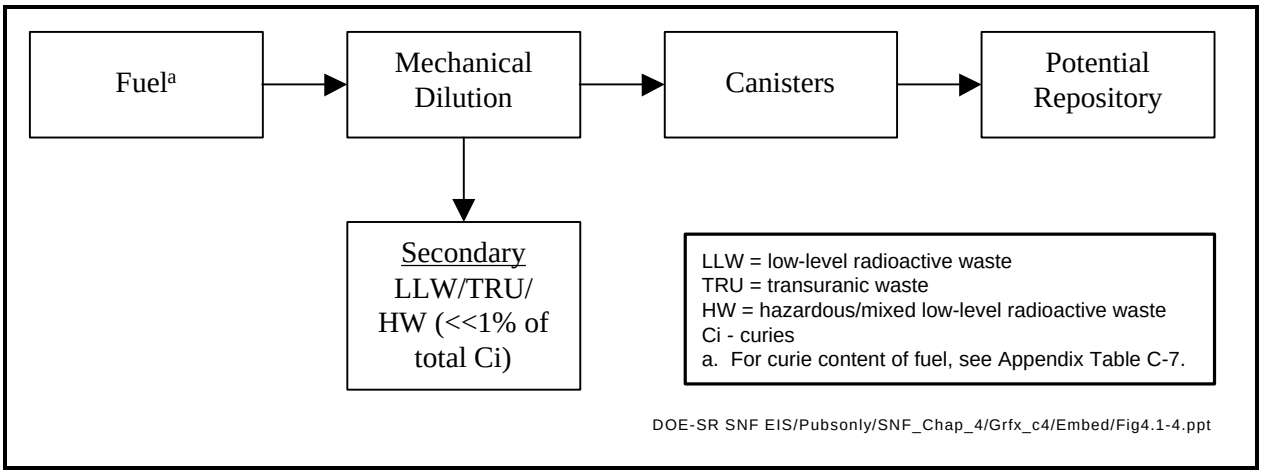

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Figure 4.1-4. Type and source of waste streams generated by the Mechanical Dilution technology option.

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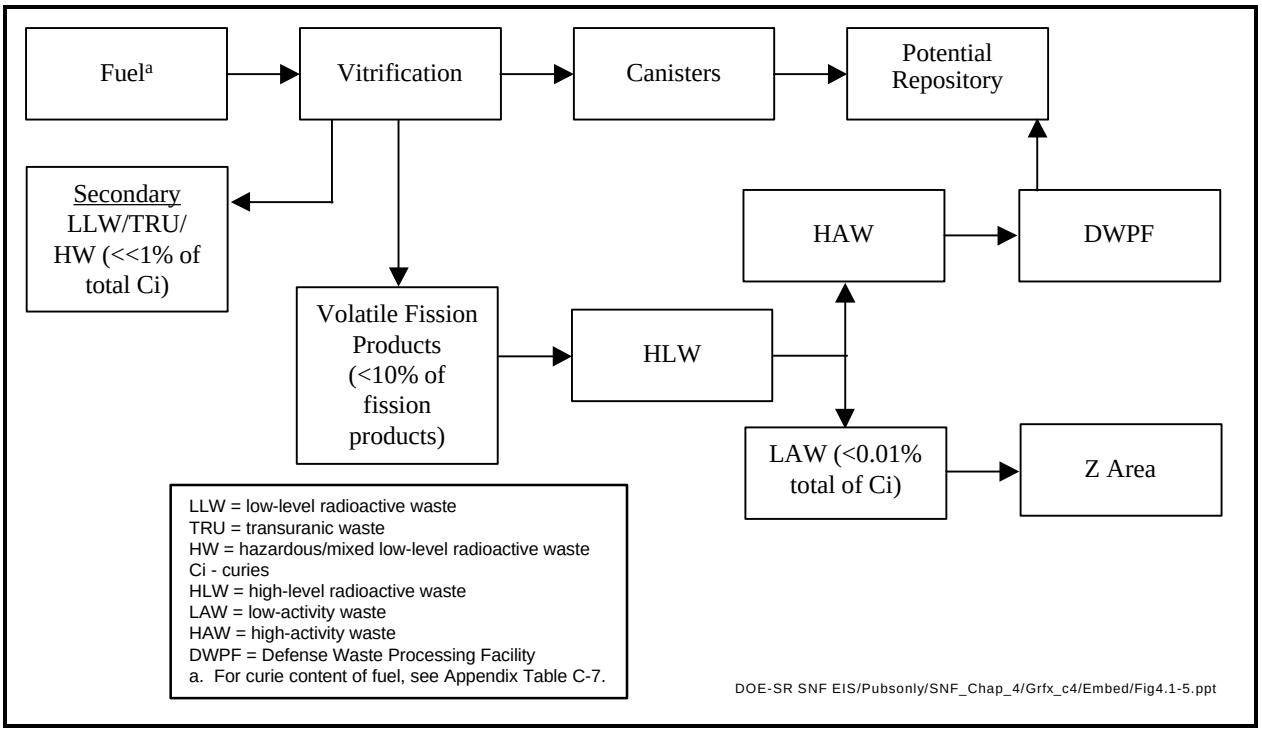

Figure 4.1-5. Type and source of waste streams generated by the Vitrification technology options.

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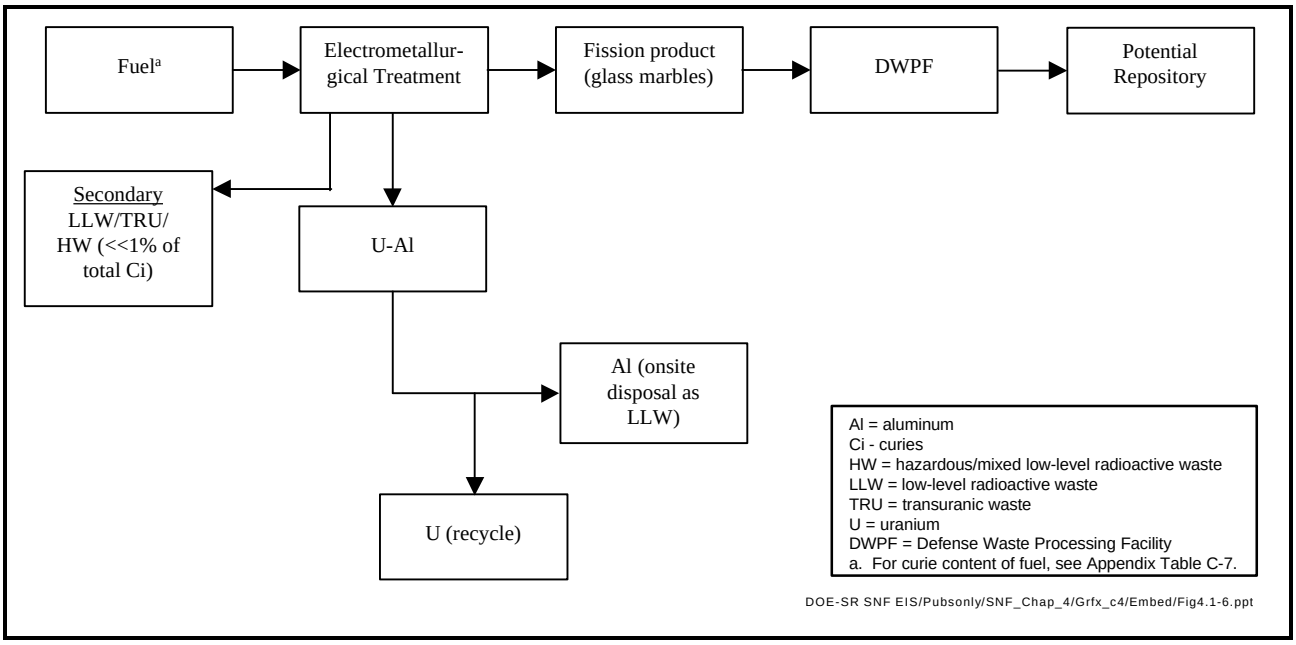

Figure 4.1-6. Type and source of waste streams generated by the Electrometallurgical Treatment technology option.

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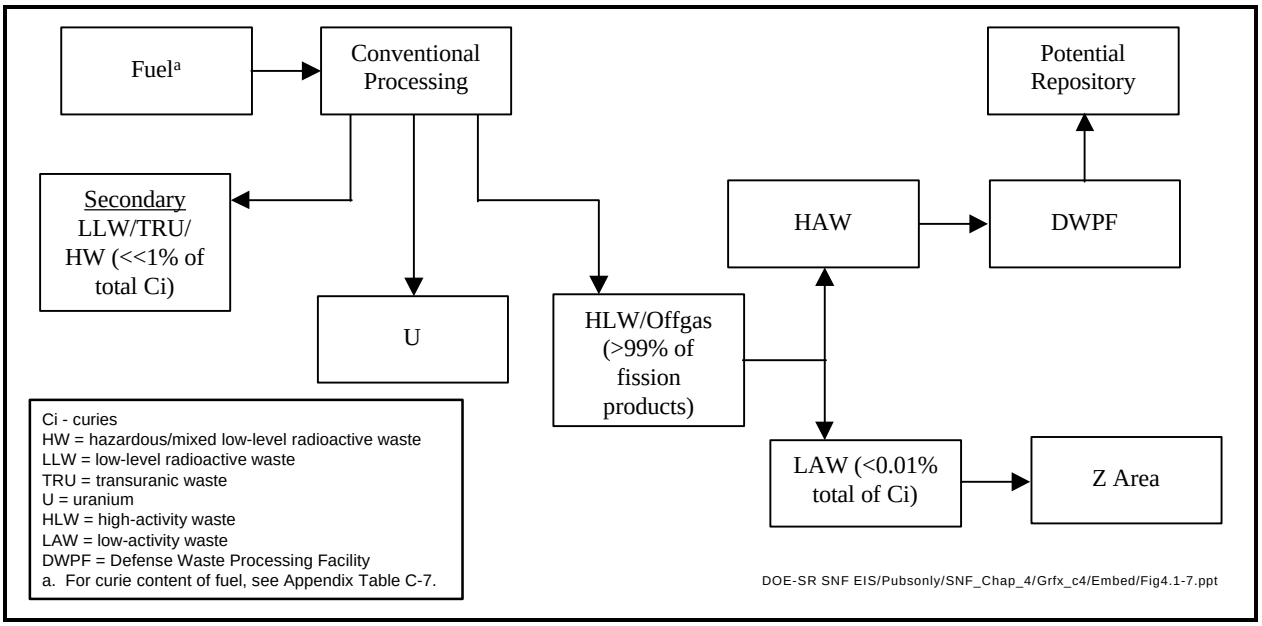

Figure 4.1-7. Type and source of waste streams generated by the Conventional Processing technology option.

rent and past SRS operations or on engineering judgments for new treatment technologies.

The following paragraphs describe estimated utility requirements for the options.

4.1.1.5.1 Water Use

Vitrification and Electrometallurgical Treatment would require the most water, followed by Conventional Processing. Total requirements for Vitrification and Electrometallurgical Treatment of all applicable fuel groups would be less than 6,000 liters per year, (the equivalent of 4.3 gallons per day) which is a minute portion (0.00001 percent) of groundwater withdrawal of more than 5×10^9 liters per year (DOE 1997). Due to the comparatively long period required to process the HEU/LEU oxides and silicides requiring resizing or special packaging (Fuel Group C) and the loose uranium oxide in cans (Fuel Group D), the Conventional Processing technology would require the greatest amount of water for those groups. For the higher actinide targets, Repackage and Prepare to Ship would require 67 percent of the water needed to support the only other option under consideration for that fuel group, Continued Wet Storage. In general, the Direct Disposal/Direct Co-Disposal, Melt and Dilute, Mechanical Dilution, and Repackage and Prepare to Ship technologies would require the least water for their applicable fuel groups, approximately 5 to 6 percent of the maximum requirement for a given group.

4.1.1.5.2 Electricity Use

Vitrification and Electrometallurgical Treatment would have the highest annual demand for electricity, followed by Conventional Processing. Differences in the time necessary to treat a fuel group under different options would affect total electricity requirements. Due to the longer period required to process the materials test reactor-like fuels (Fuel Group B), HEU/LEU oxides and silicides requiring resizing or special packaging (Fuel Group C), and loose uranium oxide in cans (Fuel Group D), Conventional Processing would require the most total electricity for those groups. For the higher actinide targets, Repackage and Prepare to Ship would require less than half the electricity needed to support continued wet storage. In general, for the appropriate fuel groups, the least electricity would be required to support Direct Co-Disposal and Mechanical Dilution.

Annually, the maximum impact alternative electrical demand is 23,600 megawatt-hours, which is approximately 3.5 percent of the current SRS annual usage of 660,000 megawatt-hours.

4.1.1.5.3 Steam Use

Where applicable, Conventional Processing would have the highest annual demand for steam. For higher actinide targets, Repackage and Prepare to Ship would require half the steam needed to support continued wet storage. In general, Direct Co-Disposal and Mechanical Dilution would require the least steam.

4.1.1.5.4 Diesel Fuel Use

For several options, DOE would use diesel fuel to support SNF treatment and storage. On an annual basis, Conventional Processing and Melt and Dilute would need the most diesel fuel. The least diesel fuel would be associated with the Vitrification and Electrometallurgical Treatment technologies, because both would require fuel only to support initial wet storage. The two options that DOE is considering for the higher actinide targets (Repackage and Prepare to Ship and Continued Wet Storage) would require comparable amounts of diesel fuel.

4.1.1.6 Environmental Justice

This section examines whether minority or low-income communities (as defined in Section 3.5.3) could receive disproportionately high and adverse human health and environmental impacts as a result of the actions described in this EIS. Even though DOE does not anticipate adverse health impacts from the options, it analyzed for the possibility of "disproportionately high and adverse human health or environmental effects on minority populations or low-income populations" (Executive Order 12898). Figures 3.5-1 and 3.5-2 show minority and low-income communities by census tract. This section discusses average radiation doses that individuals in those communities could receive and compares them to predicted doses that individuals in the other communities within the 80-kilometer- (50-mile) radius region could receive.

Figure 4.1-8 has SRS as the center of a circle with 22.5-degree sectors and concentric rings from 10 to 50 miles (16 to 80 kilometers) out from the center at 10-mile (16-kilometer) intervals. For this analysis, DOE calculated a fraction of the total population dose for each sector, laid the sector circle over the census tract map, and assigned each tract to a sector. If a tract fell in more than one sector, DOE assigned it to the sector with the largest dose value.

DOE analyzed impacts by comparing the per capita dose that each type of community would receive to doses other types of communities in the same ring would receive. To eliminate the possibility of diluting and masking impacts to a low-population community close to SRS with a high dose per person by including them with impacts to a high-population community farther from the Site, the analysis made comparisons in a series of concentric circles, the radii of which increase in 10-mile (16-kilometer) increments.

To determine the radiation dose received per person in each type of community, the analysis multiplied the number of people in each tract by that tract's dose value to obtain a total community population dose for each tract, summed these population doses in each concentric circle, and divided by the total community population in the circle to get a community per capita dose for each area of the circle. Because the per capita dose for communities (Table 4.1-19) would be constant for every alternative, the relative differences in impacts between communities would also be constant. Thus, Figure 4.1-9 and Table 4.1-19 indicate the distribution of per capita doses to types of communities in the 50-mile (80-kilometer) region. As shown in Figure 4.1-9, atmospheric releases would not disproportionately affect minority communities (population equal to or greater than 35 percent of the total population) or low income (equal to or greater than 25 percent of the total population) in the 50-mile region; that is, a comparison

Figure 4.1-8. Annular sectors around the Savannah River Site.

Table 4.1-19. Estimated per capita annual dose (rem) for identified communities in 80-kilometer (50-mile) region.[a]

	Low income		Minorities			
Distance	Less than 25 percent of population (rem)	Equal to or more than 25 percent of population (rem)	Less than 35 percent of population (rem)	35 percent to 50 percent of population (rem)	Equal to or more than 50 percent of population (rem)	All communities (rem)
0-10 miles (0-16 km[b])	1.1×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.2×10^{-5}	1.0×10^{-5}	1.1×10^{-5}
0-20 miles (0-32 km)	5.0×10^{-6}	5.0×10^{-6}	5.0×10^{-6}	7.0×10^{-6}	4.0×10^{-6}	5.0×10^{-6}
0-30 miles (0-48 km)	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-6}	2.0×10^{-6}	3.0×10^{-6}
0-40 miles (0-64 km)	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	3.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}
0-50 miles (0-80 km)	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}

a. Per capita dose based on a population dose of 1 person-rem. Per capita doses for other population doses can be obtained by multiplying the values in this table by the population dose.
b. km = kilometers.

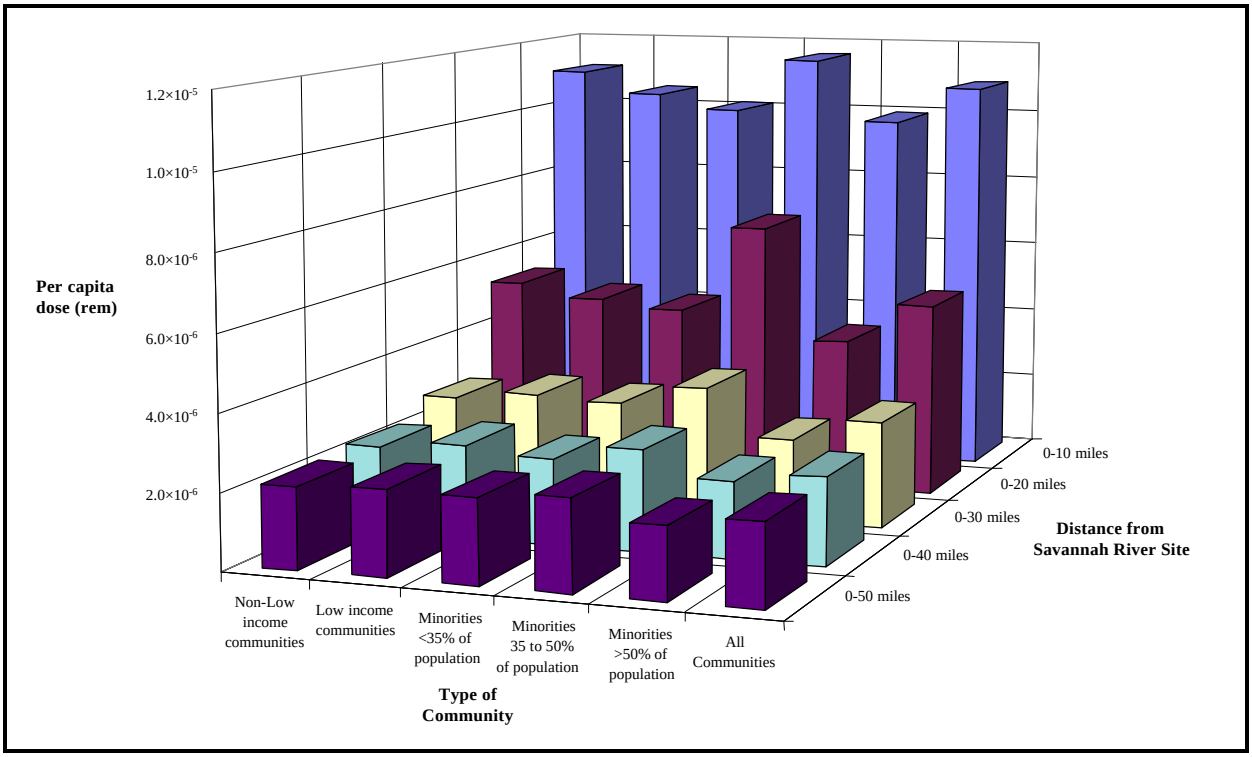

Figure 4.1-9. Distribution of a hypothetical unit population dose among SRS communities.

of per capita doses indicates that they do not vary greatly.

For example, DOE used an annual total population dose of 1 person-rem to prepare Figure 4.1-9 and its supporting data in Table 4.1-19. In comparison, the maximum annual total population dose of 0.56 person-rem for the maximum impact alternative (see Section 4.1.2) would result in 56 percent of the impact shown in Figure 4.1-9 and Table 4.1-19. For any other population dose, the per capita dose for communities can be determined by multiplying that population dose by the values listed in Table 4.1-19.

The distribution of carcinogenic and criteria pollutant emissions from routine operations and of criteria pollutants from construction activities would be essentially identical to those described for airborne radiological emissions because the distribution pathways would be the same. As a result, nonradiological emissions from any option would not cause disproportionate impacts on minority or low-income communities. Because nonradiological pollutant emissions would cause minimal impacts for any option, and because there would not be disproportionate distribution of these impacts among types of communities, environmental justice concerns would not be associated with the alternatives.

4.1.1.7 Transportation

This section discusses the potential radiological consequences of the onsite transportation of SNF and the potential consequences of transportation to a geologic repository. All onsite shipments (those that originate and terminate on SRS) would be by rail. Movements of SNF within an SRS area (e.g., H Area or F Area) are operational transfers, not onsite shipments. The potential consequences of shipping SNF from the SRS to a geologic repository are a conservative (based on worst-case number of shipments and mode of transportation) representation of impacts based on preliminary information. The full analysis of transportation impacts will be included in the EIS for a Geological Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada (currently in preparation).

4.1.1.7.1 Onsite Incident-Free Transportation Analysis [SRS]

The analysis assumed a crew of four engineers for each shipment and that the external dose rate 6.6 feet (2 meters) from the shipping cask was 100 millirem per hour (HNUS 1994a), 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 (39.4 feet [12 meters]). The duration of exposure would depend on the transport vehicle speed. In addition, vehicle crew time would depend on the distance of each shipment.

Table 4.1-20 summarizes the collective doses (person-rem) and health effects (latent cancer fatalities) associated with a single incident-free onsite shipment of SNF at SRS.

To determine the incident-free transportation dose for management of all SRS spent nuclear fuel, it is necessary to calculate the total dose over all shipments. DOE has estimated that it would take approximately 150 rail shipments to de-inventory the Receiving Basin for Offsite Fuels to the L-Area Disassembly Basin. This action would occur under all alternatives, including the No-Action Alternative. The radiation dose to the crew from these shipments is estimated to be approximately 0.57 person-rem, which could result in 2.3×10^{-4} latent cancer fatalities.

DOE has estimated that it would take approximately 300 rail shipments to transport the contents of the L-Area Disassembly Basin (including the fuel that was previously in the Receiving Basin for Offsite Fuels) to the Transfer and Storage Facility; the Transfer, Storage, and Treatment Facility; or the F- and H-Area Canyons. This action would occur under all alternatives, except the No-Action Alternative. Assuming the bounding location for the

Table 4.1-20. Collective doses and health effects for onsite incident-free SNF shipments.[a]

Shipment origin/destination	Crew dose per shipment (person-rem)	Number of LCFs[b] per shipment
		Crew
L Area/H Area	3.80×10^{-3}	1.52×10^{-6}
L Area/F Area	4.10×10^{-3}	1.64×10^{-6}
F Area/H Area	1.40×10^{-3}	5.60×10^{-7}
P Area/H Area	4.90×10^{-3}	1.96×10^{-6}
P Area/F Area	3.88×10^{-3}	1.55×10^{-6}
C Area/H Area	3.33×10^{-3}	1.33×10^{-6}
C Area/F Area	4.20×10^{-3}	1.68×10^{-6}

a. Derived from HNUS (1994a).
b. LCF = latent cancer fatality.

Transfer and Storage Facility or the Transfer, Storage, and Treatment Facility, the radiation dose to the crew from these shipments is estimated to be approximately 1.23 person-rem which could result in $4.9\text{x}10^{-4}$ latent cancer fatalities. Therefore, for the No-Action Alternative, the total radiation dose to the shipping crew would be approximately 0.57 person-rem, which could result in $2.3\text{x}10^{-4}$ latent cancer fatalities. For all other alternatives, the total radiation dose to the crew would be approximately 1.8 person-rem, which could result in $7.2\text{x}10^{-4}$ latent cancer fatalities.

4.1.1.7.2 Incident-Free Transportation Analysis [Geologic Repository]

DOE estimated the impacts of shipping SNF from SRS to a theoretical geologic repository in the Western United States (approximately 4,000 kilometers [2,500 miles] from SRS) by truck. TC This analysis assumes all shipments from SRS, approximately 1,400 (worst case among the alternatives), would be by truck because the impacts would bound the impacts of rail shipments. Because the transport of SRS spent fuel would use existing highways, it would represent a very small fraction of national highway traffic. Consequently, there would be negligible impacts on land use; air quality; hydrology; biological resources and cultural resources; socioeconomics; noise; aesthetics; utilities, energy, and materials; or waste management. The analysis of the potential impacts of transporting SRS spent nuclear fuel to the repository focuses on the potential radiological impacts to workers and the public.

DOE recognizes that it cannot predict with any certainty the specific routes that would be used to ship SNF to a repository. Nonetheless, the analysis uses current regulations governing highway shipments to select actual highway routes to estimate the potential environmental impacts of national transportation. TC Assumed distances within the various rural, suburban, and urban population zones can be found on Table 4.1-21.

Loading Operations

Prior to shipping the fuel, DOE would load it into NRC certified Type B shipping casks. The potential dose to involved workers from the loading operation would be less than that expected at a commercial nuclear facility because the radionuclide inventory of commercial fuel is higher than that of the DOE SNF. The dose would be further limited by worker rotation and other administrative controls. DOE expects any dose to uninvolved workers would be negligible because they would not have tasks that could result in radiation exposure. Likewise, DOE expects radiation exposure to the public would not occur because of the distance of the loading operations from the areas of public access.

Table 4.1-21. Incident-free radiological impacts of 1,400 offsite truck shipments of spent nuclear fuel to the proposed Yucca Mountain Geologic Repository.

Exposure group	Unit risk factors (person-rem kilometer)[a]			Kilometers traveled		
	Rural	Suburban	Urban	Rural	Suburban	Urban
Occupational	4.6×10^{-5}	1.0×10^{-4}	1.7×10^{-4}	3,292.6	570.2	65.9
Off-link[b]	1.2×10^{-7}	1.6×10^{-5}	1.1×10^{-4}	3,292.6	570.2	65.9
On-link[c]	5.0×10^{-6}	1.5×10^{-5}	1.5×10^{-4}	3,292.6	570.2	65.9
Stops	1.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}	3,292.6	570.2	65.9

	Collective dose (person-rem)			Total collective dose	LCF[d]
	Rural	Suburban	Urban		
Occupational	212	80	16	308	0.123
General population					
Off-link[b]	1	13	10	24	0.012
On-link[c]	23	12	14	49	0.024
Stops	553	96	11	660	0.330
General population total					0.366

a. The methodology, equations, and data used to develop the unit risk factors are discussed in Madsen et al. (1986) and Neuhauser and Kanipe (1992). Cashwell et al. (1986) contains a detailed explanation of the use of unit risk factors.
b. Off-link general population are persons within 800 meters (2,625 feet) of the highway.
c. On-link general population are persons sharing the highway.
d. LCF = latent cancer fatality.

Transportation to a Geologic Repository

To estimate the potential impacts of incident-free transportation of SNF to a repository, the analysis considered both the public and workers. Unit risk factors commonly used in a number of other DOE EISs were used to determine the potential person-rem exposure per kilometer for both workers and public. In the case of the general population, both off-link and on-link doses were calculated. The off-link dose could affect persons within 800 meters (2,625 feet) of the highway; the on-link dose could affect persons sharing the highway. Table 4.1-21 presents the potential incident-free radiological impacts from 1,400 shipments of SNF from the SRS to a theoretical geologic repository. As can be seen from the table, potential latent cancer fatalities could result in less than 1 additional death from radiation over the life of the shipments.

4.1.1.7.3 Onsite Transportation Accident Analysis [SRS]

DOE analyzed radiological impacts from potential accidents to the onsite maximally exposed individual from onsite rail shipments. The analysis calculated doses using the RADTRAN computer code (Neuhauser and Kanipe 1992) with site-specific meteorology, and calculated risk using site-specific rail accident rates and accident probabilities (HNUS 1994b).

The analysis assumed a release of the maximum reasonably foreseeable amount of radioactive material for the type of SNF shipped on SRS (HNUS 1994b). Radiological doses were modeled for three human receptor groups: the onsite worker population, members of the public residing near SRS, and the maximally exposed offsite individual. The consequences are expressed as excess latent cancer fatalities in each receptor group.

Table 4.1-22 summarizes the radiation doses resulting from the most severe reasonably foreseeable onsite transportation accident and associated latent cancer fatalities.

4.1.1.7.4 Transportation Accident Analysis [Geologic Repository]

EC Potential impacts from accidents resulting from transporting SNF to a geologic repository are not quantified in this document but have been analyzed in the EIS for a Geologic Repository for Disposal of Spent Nuclear Fuel and High-Level EC Radioactive Waste at Yucca Mountain, Nye County, Nevada. Previous EISs, including the Foreign Research Reactor Spent Fuel EIS (DOE 1996) and the Programmatic Spent Fuel EIS (DOE 1995b) analyzed the potential accident impacts of transporting SNF. The following discussions summarize the types of accidents that could be expected. Impacts are presented in Table 4.1-23.

Loading Operation

In general, accidents from loading operations could be caused by unplanned contact (bumping) during lifting or handling of casks, canisters, or fuel assemblies. Initiating events could include fires, explosions, earthquakes, cask tor nadoes, canister or basket drops, and loaded shipping drops. The Interim Management of Nuclear Materials at SRS EIS (DOE 1995a) assessed the radiological impacts from potential accidents associated with preparing, storing, and onsite shipment of some spent nuclear fuel.

Transportation to a Geologic Repository

Several types of accidents potentially could occur while transporting SNF. The first type of accident, resulting in the most radiological exposure to the public, assumes the breach of a shipping cask during an accident resulting in the release of a fraction of its contents to the air. This accident would be very unlikely. The second type of accident would involve truck wrecks that could result in non-radiological fatalities to workers or members of the public. The probability of an accident is dependent upon the number of shipments made and total miles traveled. EC

4.1.2 IMPACTS OF THE ALTERNATIVES

As discussed in Chapter 2, none of the options for the management of SNF, except Continued Wet Storage, would address the requirements of all six fuel types. Therefore, DOE must consider combinations of technologies to satisfy the purpose and need identified in Chapter 1. This

Table 4.1-22. Impacts on SRS workers, maximally exposed offsite individuals, and offsite population from SNF transportation accidents on Savannah River Site.

Accident frequency	Worker dose (rem)	Probability of a worker LCF[b]	MEI[c] dose (rem)	Probability of a LCF to the MEI	Population dose (person-rem)	Population LCFs
1.28×10^{-4}	2.78	1.11×10^{-3}	2.2×10^{-5}	1.08×10^{-8}	0.16	8.21×10^{-5}

a. Source: DOE (1995a).
b. LCF = latent cancer fatality.
TC c. MEI = maximally exposed individual.

Table 4.1-23. Truck transportation accident analysis impacts.

Radiological impacts				Traffic impacts		
Risk factor (person-rem/ shipment)[a]	Maximum number shipments	Total (person-rem)	Total LCFs	Risk factor (fatality/ shipment)[b]	Maximum number shipments	Total fatality
TC 1.79×10^{-5}	1,400	0.025	1.25×10^{-5}	1.12×10^{-4}	1,400	0.16

LCF = latent cancer fatalities.
a. DOE (1996).
TC b. Adapted from DOE (1999).

section provides the results of analyzing combinations of the technology options applicable to the fuel groups. Excluding continued wet storage, there are more than 700 combinations of technology options and fuel groups that could be analyzed. However, it would be impractical and unreasonable to do so. DOE has identified four sets of combinations for analysis as alternatives in this EIS (in addition to No Action) which it believes are representative. These four alternatives are the Minimum Impact Alternative, Direct Disposal Alternative, Preferred Alternative, and Maximum Impact Alternative. The data in Section 4.1.1 can be used to compile the impacts of other configurations of viable cases.

Continued wet storage for all fuel types is the No-Action Alternative. National Environmental Policy Act (NEPA) regulations require the evaluation of No Action, (which would not meet the purpose and need described in Chapter 1); however, it provides a baseline against which DOE can compare the action alternative combinations.

The second alternative, Minimum Impact, would result in the smallest environmental impacts to human health. It is also the environmentally-preferred alternative.

The third alternative is Direct Disposal. All fuel types that could be dry stored would be. Higher Actinide Targets and Non-Aluminum-Clad Fuels would be Repackaged and Prepared to Ship Offsite. Uranium and Thorium Metal Fuels and Loose Uranium Oxide in Cans would undergo conventional processing.

The fourth alternative is the Preferred Alternative. Melt and Dilute would be used to treat the Materials Test Reactor-like fuels, most of the HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (Group C), and most of the Loose Uranium Oxide in Cans (Group D). Group A and the remaining Group C and Group D fuels (<10 percent of the material in these fuel groups) would be treated with conventional processing. Finally, the Higher Actinide Targets and the Non-Aluminum-Clad fuels would be Repackaged and Prepared to Ship offsite.

EC

The final alternative would apply the chemical processing option to all the fuel except the higher actinide targets and non-aluminum-clad SNF and probably would produce the greatest environmental impacts, and therefore, provides an upper bound. It is termed the Maximum Impact Alternative. Section 2.4 provides a complete description of the SNF management alternatives.

Tables 4.1-24 through 4.1-26 list the impacts of the five alternatives summed from the operational impacts of each appropriate technology presented in Section 4.1.1. The following sections describe the alternatives and the bases for their selection. The conclusions from Section 4.1.1.5 on environmental justice would apply to all the alternatives.

DOE based the values listed for annual radiation dose to the noninvolved worker, the offsite maximally exposed individual, and the 620,000-person population surrounding SRS on the sum of the annual doses for each technology-fuel group included in the alternative. Since the time intervals over which these annual doses would occur might not coincide, this method could overestimate the annual doses that actually would occur.

The values in Table 4.1-26 for health effects to the noninvolved worker, maximally exposed individual, and the offsite population for the No-Action Alternative represent current reactor area emissions (including two SNF wet basins) for the entire period of analysis. The values for the other alternatives would be incremental above these baseline values. Summing these baseline and incremental values would be conservative, however, because there would not be two SNF wet basins operating over the entire 38-year period of analysis.

Table 4.1-24. Estimated maximum incremental concentrations of nonradiological air pollutants for the noninvolved worker.

Pollutant	Averaging Time	Regulatory Standard[a]	No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative
Toxic Pollutants (mg/m^3)							
Nitric acid	24-hour	5	0.03	0.02	2.75	2.62	7.95
1,1,1-Trichloroethane	24-hour	1,900	–	–	0.02	0.02	0.05
Benzene	24-hour	3.19	–	–	0.02	0.02	0.05
Ethanolamine	24-hour	6	0.03	0.02	0.02	0.02	0.03
Ethyl benzene	24-hour	435	–	–	0.01	0.01	0.02
Ethylene glycol	24-hour	None	0.03	0.02	0.02	0.02	0.03
Formaldehyde	24-hour	0.75	0.03	0.02	0.02	0.02	0.03
Glycol ethers	24-hour	80	0.03	0.02	0.02	0.02	0.03
Hexachloronaphthalene	24-hour	0.2	0.03	0.02	0.02	0.02	0.03
Hexane	24-hour	1,800	0.03	0.02	0.03	0.03	0.06
Manganese	24-hour	5	–	–	0.01	0.01	0.02
Mercury	24-hour	0.1	–	–	0.01	0.01	0.02
Methyl alcohol	24-hour	260	0.03	0.02	0.02	0.02	0.03
Methyl ethyl ketone	24-hour	590	0.03	0.02	0.02	0.02	0.03
Methyl isobutyl ketone	24-hour	410	–	–	0.01	0.01	0.02
Methylene chloride	24-hour	86.7	–	–	0.02	0.02	0.05
Napthalene	24-hour	50	0.03	0.02	0.02	0.02	0.03
Phenol	24-hour	19	–	–	0.01	0.01	0.02
Phosphorus	24-hour	0.1	–	–	0.01	0.01	0.02
Sodium hydroxide	24-hour	2.0	–	–	0.01	0.01	0.02
Toluene	24-hour	754	0.03	0.02	0.03	0.03	0.06
Trichloroethene	24-hour	537	–	–	0.01	0.01	0.02
Vinyl acetate	24-hour	None	–	–	0.01	0.01	0.02
Xylene	24-hour	435	0.03	0.02	0.05	0.05	0.10
Criteria Pollutants ($\mu g/m^3$)							
Nitrogen oxides	Annual	NA	–	0.05	38.2	36.4	111
Total Suspended Particulates (total dust)	8-hour	15	–	0.02	0.35	0.34	0.99
Particulate Matter (<10 μm)	8-hour	5	–	0.09	0.08	0.08	0.05
	24-hour	NA	–	0.99	0.86	0.87	0.62
Carbon monoxide	8-hour	55	0.03	0.25	1.81	1.82	4.78
	1-hour	NA	0.03	0.79	5.65	5.68	14.93
Sulfur dioxide	Annual	NA	–	0.02	0.04	0.04	0.08
	8-hour	13	–	0.02	0.31	0.30	0.86
	3-hour	NA	–	0.02	0.72	0.70	2.07
Gaseous fluorides	1-month	None	–	-	0.10	0.10	0.29
	1-week	NA	–	-	0.18	0.17	0.52
	24-hour	NA	–	-	0.55	0.52	1.59
	12-hour	NA	–	-	0.80	0.76	2.32
Ozone (as VOC)	1-hour	0.2	–	nc	nc	nc	nc

– = no air emission associated with this combination.
NA = not applicable.
nc = not calculated.
VOC = volatile organic compound.
a. 29 CFR 1910.1000, Subpart Z and OSHA 8-hour time-weighted averages.

Table 4.1-25. Estimated maximum incremental concentrations of nonradiological air pollutants at the Site boundary.

Pollutant	Averaging Time	Regulatory Standard[a]	No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative	Maximum Impact Alternative	
Toxic Pollutants (mg/m^3)								
Nitric acid	24-hour	125	–	–	0.11	0.10	0.31	
1,1,1-Trichloroethane	24-hour	9,550	0.03	0.03	0.03	0.03	0.03	
Benzene	24-hour	150	–	–	0.01	0.01	0.02	
Ethanolamine	24-hour	200	0.03	0.03	0.03	0.03	0.03	
Ethyl benzene	24-hour	4,350	–	–	0.01	0.01	0.02	
Ethylene glycol	24-hour	650	0.03	0.03	0.03	0.03	0.03	
Formaldehyde	24-hour	15	0.03	0.03	0.03	0.03	0.03	
Glycol ethers	24-hour	+	0.03	0.03	0.03	0.03	0.03	
Hexachloronaphthalene	24-hour	1	0.03	0.03	0.03	0.03	0.03	
Hexane	24-hour	200	0.03	0.03	0.03	0.03	0.03	TC
Manganese	24-hour	25	–	–	0.01	0.01	0.02	
Mercury	24-hour	0.25	–	–	0.01	0.01	0.02	
Methyl alcohol	24-hour	1,310	0.03	0.03	0.03	0.03	0.03	
Methyl ethyl ketone	24-hour	14,750	0.03	0.03	0.03	0.03	0.03	
Methyl isobutyl ketone	24-hour	2,050	–	–	0.01	0.01	0.02	
Methylene chloride	24-hour	8,750	–	–	0.01	0.01	0.02	
Napthalene	24-hour	1,250	0.03	0.03	0.03	0.03	0.03	
Phenol	24-hour	190	–	–	0.01	0.01	0.02	
Phosphorus	24-hour	0.5	–	–	0.01	0.01	0.02	
Sodium hydroxide	24-hour	20	–	–	0.01	0.01	0.02	TC
Toluene	24-hour	2,000	0.03	0.03	0.03	0.03	0.03	
Trichloroethene	24-hour	6,750	–	–	0.01	0.01	0.02	
Vinyl acetate	24-hour	176	–	–	0.01	0.01	0.02	
Xylene	24-hour	4,350	0.03	0.03	0.03	0.03	0.03	
Criteria Pollutants ($\mu g/m^3$)								
Nitrogen oxide	Annual	100	0.03	0.02	1.17	1.12	3.36	
Total Suspended Particulates	Annual	75	0.03	0.02	0.02	0.02	0.02	
Particulate Matter (<10 µm)	Annual	50	–	–	0.01	0.01	0.02	
	24-hour	150	–	–	0.05	0.04	0.13	
Carbon monoxide	8-hours	10,000	0.03	0.07	0.49	0.50	1.31	
	1-hour	40,000	0.03	0.37	3.60	3.57	9.76	
Sulfur dioxide	Annual	80	–	0.02	0.02	0.02	0.02	
	24-hour	365	–	0.03	0.07	0.07	0.13	
	3-hour	1300	–	–	0.34	0.32	0.98	
Gaseous fluoride	1-month	0.8	–	–	0.01	0.01	0.02	
	1-week	1.6	–	–	0.02	0.01	0.04	
	24-hour	2.9	–	–	0.03	0.02	0.07	
	12-hour	3.7	–	–	0.05	0.04	0.13	
Ozone (as VOC)	1-hour	235	–	0.16	0.38	0.41	0.80	

– = no air emission associated with this option.
+ = no state standard.
VOC = volatile organic compound.
a. SCDHEC standard No. 2 (criteria pollutants) and No. 8 (toxic pollutants). TC

Table 4.1-26. Impacts from alternatives.[a]

Impact	No Action Alternative	Minimum Impact Alternative	Direct Disposal Alternative	Preferred Alternative[b]	Maximum Impact Alternative
Health Effects for the Entire Period of Analysis (1998-2035)[f]					
MEI[c] dose (millirem)	0.63[d]	6.1×10^{-4}	7.2×10^{-3}	0.19	0.67
MEI LCF[e] probability	3.1×10^{-7}[d]	3.0×10^{-10}	3.6×10^{-9}	9.5×10^{-8}	3.4×10^{-7}
Population dose (person-rem)	22.6[d]	0.022	0.077	6.9	8.7
Population LCFs (unitless)	0.011[d]	1.1×10^{-5}	3.8×10^{-5}	3.4×10^{-3}	4.4×10^{-3}
Collective worker dose (person-rem)	760	690	840	841	2,100
Collective worker LCFs (unitless)	0.30	0.28	0.34	0.33	0.84
Noninvolved worker dose (millirem)	4.25[d]	5.0×10^{-3}	0.02	1.53	1.53
Noninvolved worker LCF probability	1.7×10^{-6}[d]	2.0×10^{-9}	9.6×10^{-9}	6.1×10^{-7}	6.3×10^{-7}
Annual Radiological Air Emission Impacts					
Maximum annual MEI[d] dose (millirem)	0.02[d]	6.1×10^{-4}	7.4×10^{-4}	0.044	0.015
Maximum annual population dose (person-rem)	0.59[d]	0.022	0.027	1.6	0.56
Maximum annual noninvolved worker dose (millirem)	0.11[d]	5.0×10^{-3}	6.0×10^{-3}	0.36	0.12
Annual Radiological Liquid Emission Impacts					
Maximum annual MEI dose (millirem)	0	0	1.4×10^{-3}	4.2×10^{-5}	0.057
Maximum annual population dose (person-rem)	0	0	4.9×10^{-3}	2.4×10^{-4}	0.19
Waste Generation (cubic meters) for the Entire Period of Analysis (1998-2035)					
High-level waste					
Liquid	2,300	660	1,200	1,050	10,500
Equivalent DWPF canisters	38	11	20	17	160
Saltstone	6,100	1,800	3,200	2,700	27,000
Transuranic waste	0	15	360	563	3,700
Hazardous/low-level mixed waste	76	25	46	103	267
Low-level waste	57,000	20,000	31,000	35,260	140,000
Utilities and Energy Required for the Entire Period of Analysis (1998-2035)					
Water (millions of liters)	1,100	660	1,400	1186	8,000
Electricity (megawatt-hours)	46,000	27,000	81,000	116,000	600,000
Steam (millions of kilograms)	340	195	520	650	3,600
Diesel fuel (thousands of liters)	230	180	2,300	2760	22,000

a. In the event that fuel receipts are less than those reported in Chapter 1, the values in this table that report impacts over the entire period of analysis would be less. Instructions for scaling impacts are provided in the appropriate Chapter 4 tables that provide input to this table.
b. In the calculation of preferred alternative impacts, all the HEU/LEU oxides and silicides requiring resizing or special packaging have been accounted for in the melt and dilute technology even though a very small percentage would be conventionally processed. On the other hand, the loose-uranium-oxide-in-cans preferred alternative impacts do consider that 60 percent would be conventionally processed and the remaining 40 percent would be melted and diluted.
c. MEI = maximally exposed offsite individual.
d. Reflects current reactor-area emissions (including two SNF wet basins).
e. LCF = latent cancer fatality.
f. To calculate an annual impact, divide a number by 38. To calculate an impact for a given duration, multiply the annual impact by the duration in years. For example, the annual dose to the MEI from the preferred alternative would be 0.005 mrem (0.17/38). The estimated dose to the MEI until a storage facility would be operational (18 years from now) would be 0.040 mrem (0.005x8).

4.1.2.1 No-Action Alternative

Under the No-Action Alternative, SRS would continue to receive shipments of SNF from foreign research reactors, domestic research reactors, and other DOE sites. DOE would store the fuel in the L-Reactor Disassembly Basin or the Receiving Basin for Offsite Fuels, in addition to the currently stored SNF, under continued wet storage, and would ship the non-aluminum-clad fuel from these basins offsite. DOE would maintain the wet storage basins, performing upgrades as necessary to maintain proper water quality. The continued long-term underwater storage of aluminum-based SNF could lead to increased corrosion with increased environmental, health, and safety vulnerabilities. The No-Action Alternative consists of cases A8, B8, C8, D8, E8, and F8 (Table 4.1-27).

4.1.2.2 Minimum Impact Alternative

The identification of the Minimum Impact Alternative required both quantitative and quantitative analyses. The first step identified the minimum-impact technology for each fuel group for each analytical parameter (e.g., volume of high-level waste, air concentrations). However, the selection process often resulted in a combination of high and low impacts among parameters for a specific fuel group-technology combination cases; in other words, no clearly identified "best" or "worst" configuration was identified. Therefore, the second step was a qualitative examination of trends in configurations of cases that identified overall minimum impacts. Human health effects and environmental pollution impacts received slightly greater weight than consumption of natural resources or waste disposal space. In addition, impacts to the general public received slightly greater weight than those to SRS workers. The analysis indicates that cases A1, B1, C1, D3, E2, and F2 would provide minimum impacts (Table 4.1-28). Although other analysts could select different cases, DOE believes that the range of impacts from reasonable choices of minimum-impact scenarios would be small and that the impacts of this combination would be representative of the lower bound of impacts from the proposed action.

4.1.2.3 Direct Disposal Alternative

This alternative combines the New Packaging and the Conventional Processing Technologies. Materials Test Reactor-like fuels and HEU/LEU Oxides and Silicides (except the failed and sectioned fuels) would be treated using the Direct Disposal/Direct Co-Disposal technology and placed in the Transfer and Storage Facility with a minimum of treatment (e.g., cold-vacuum drying and canning). The repackaging of the higher actinide targets and non-aluminum-clad fuels in the Transfer and Storage Facility would use the Repackage and Prepare to Ship technology. The uranium and thorium metal fuel, loose uranium oxide in cans, and failed and sectioned fuel from the HEU/LEU Oxides and Silicides fuel group would be treated using the Conventional Processing Alternative to alleviate the potential health and safety vulnerabilities discussed in Section 2.4.3.2 and because this material probably would not be suitable for placement in a geologic repository if treated with the Direct Disposal/Co-Disposal option. Therefore, the Direct Disposal alternative consists of cases A7, B1, C1, D7, E2, and F2 (Table 4.1-29). TC

4.1.2.4 Preferred Alternative

DOE proposes to implement several of the technologies identified in Section 2.2 to manage spent nuclear fuel at SRS. These technologies are Melt and Dilute, Conventional Processing, and Repackage and Prepare to Ship. Each of these technologies would treat specific groups of spent nuclear fuel, as described below. The technology and fuel group combinations form DOE's Preferred Alternative in this EIS. The configuration of this preferred alternative is identified in Table 4.1-30.

4.1.2.4.1 Melt And Dilute

DOE has identified the Melt and Dilute process as the preferred method of treating most (about 97 percent by volume or about 32,000 MTRE) of the aluminum-based SNF considered in this EIS. DOE will continue to pursue a research and development program leading to a demonstration of the technology in FY 2001 using full-size irradiated research reactor spent nuclear fuel assemblies. With a successful demonstration of the technology, DOE expects to have ready a treatment facility to perform production melt and dilute operations in FY 2008. DOE will ensure the continued availability of SRS conventional processing facilities until we have successfully demonstrated implementation of the Melt and Dilute treatment technology.

The fuel proposed for the preferred Melt and Dilute technology includes the Material Test Reactor-like fuel, most of the Loose Uranium Oxide in Cans fuel, and most of the HEU/LEU Oxide and Silicide fuel. Exceptions are the uranium and thorium fuel, failed and sectioned oxide and silicide fuel, some loose uranium oxide in cans fuel, the Higher Actinide Targets, and non-aluminum-clad fuel.

If DOE identifies any health or safety concerns involving any aluminum-based SNF prior to the melt and dilute facility becoming operational, DOE could use F and H Canyons to stabilize the material of concern, if the canyons were not decommissioned.

4.1.2.4.2 Conventional Processing

DOE has identified conventional processing to manage a relatively small volume of aluminum-based SNF at the SRS (about 3 percent by volume; less than 3,000 MTRE) that presents a potential health and safety vulnerability or is in a form that may be unacceptable for placement in a geologic repository. That SNF includes the Experimental Breeder Reactor-II fuel, the Sodium Reactor Experiment fuel, the Mark-42 targets and the core filter block from the Uranium and Thorium Metal fuel group; the failed or sectioned Tower Shielding Reactor, High Flux Isotope Reactor, Oak Ridge Reactor, and Heavy Water Components Test Reactor fuels and a Mark-14 target from the HEU/LEU Oxides and Silicides fuel group; and the Sterling Forest Oxide (and any other powdered/oxide fuel that may be received at SRS while H Canyon is still in operation) from the Loose Uranium Oxide in Cans fuel group.

4.1.2.4.3 Repackaging

DOE proposes to repackage the non-aluminum-clad fuel at SRS and transfer the material to dry storage. DOE would transfer the non-aluminum-clad fuel to that facility for storage pending offsite shipment. DOE expects transfer operations would begin in time to support closing the Receiving Basin for Offsite Fuels by 2007. Depending on receipt schedules for research reactor fuels and the operating schedule for the melt and dilute facility, DOE could deinventory the Receiving Basin for Offsite Fuels and move any remain fuel to the Building 105-L wet basin prior to packaging the fuel for dry storage.

The Preferred Alternative would include cases A7, B3, C3, D3, E2, and F2 (Table 4.1-30).

4.1.2.4.4 Continued Wet Storage

DOE proposed to maintain the higher actinide target fuel group in continued wet storage pending decisions on final dispositon.

4.1.2.5 Maximum Impact Alternative

This alternative provides the upper bound on the range of impacts from potential configurations. It would provide conventional processing for all SNF except the higher actinide targets and the non-aluminum-clad fuels selected for offsite shipment and deemed inappropriate for conventional processing. The higher actinide targets would be repackaged for potential offsite shipment and dry-stored until DOE made a decision regarding their disposition. The non-aluminum-clad fuels would be packaged for shipment and dry stored until they were ready for shipment to the Idaho National Engineering and Environmental Laboratory.

Analyses of the maximum impact alternative are conservative in that they assume that the entire SNF inventory would be processed in the canyons, which would produce the greatest impacts of all the treatment options. No credit is taken for discontinuing use of the canyons and processing some of the inventory in a new treatment facility. The Conventional Processing Alternative would include cases A7, B7, C7, D7, E2, and F2 (Table 4.1-31). DOE believes that this combination would provide an upper bound on impacts.

4.2 Accident Analysis

This section summarizes risks to the public and workers from potential accidents associated with the technology options for SNF management at the SRS.

An accident is a sequence of one or more unplanned events with potential outcomes that endanger the health and safety of workers and the public. An accident can involve a combined release of energy and hazardous materials (radiological or chemical) that might cause prompt or latent health effects. The sequence usually begins with an initiating event, such as a human error followed by an explosion, or an earthquake followed by structural failure. A succession of other events, such as a ventilation system failure, that are dependent or independent of the initial event, could affect the magnitude of the accident and the materials released. Initiating events fall into three categories:

- *Internal initiators* normally originate in and around the facility but are always a result of facility operations (equipment or structural failures, human errors, internal flooding).

- *External initiators* are independent of facility operations and normally originate outside the facility (aircraft crashes, nearby explosions, and toxic chemical releases at nearby facilities that affect worker performance); some can affect the ability of the facility to maintain confinement of hazardous materials because of structural damage.

- *Natural* phenomena initiators are natural occurrences that are independent of facility operations and of *events* at nearby facilities or operations (earthquakes, high winds, floods, lightning, snow). Natural phenomena initiators could affect external facilities, which could in turn affect other facilities and compound the progression of the accident.

Table 4.2-1 summarizes the estimated impacts to workers and the public from potential accidents for each SNF technology option. All the options would require the use of the Receiving Basin for Offsite Fuels and the L-Reactor Disassembly Basin. All except Continued Wet Storage would require the construction and operation of a Transfer and Storage Facility or a Transfer, Storage, and Treatment Facility.

The table lists the impacts of potential accidents in relation to the phases required to implement each option. They list only the accident with the worst impacts based on the maximally exposed offsite individual. Appendix D contains details of the impacts of other postulated accidents. Table 4.2-1 lists potential accident consequences as latent cancer fatalities, without consideration of the accident's probability. The calculation of latent cancer fatalities from population dose is performed in the same manner as for non-accident radiological health effects presented in section 4.1.1.3.1. TC

DOE estimated impacts to three receptors: (1) an uninvolved worker 2,100 feet (640 meters) from the accident location as discussed in DOE (1994), (2) the maximally exposed individual at the SRS boundary, and (3) the offsite population in an area within 50 miles (80 kilometers).

Many of the analysis results presented in Table 4.2-1 are substantially different from those given in the draft EIS. DOE has continued to conduct research and development, including accident analyses, to determine the feasibility of implementing technologies and the potential health and safety consequences of doing so. In some cases design changes have been TC

Table 4.2-1. Estimated maximum consequence accident for each technology.

Option	Accident Frequency	Consequences			
		Noninvolved Worker (rem)	MEI (rem)	Offsite Population (person-rem)	Latent Cancer Fatalities
Continued Wet Storage (No Action)[a]					
RBOF (high wind-induced criticality)	Once in 26,000 years	13	0.22	12,000	6.2
L-Reactor basin (basin-water draindown)	Once in 500 years	0.014	0.016	(b)	(b)
Direct Co-Disposal					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Repackage and Prepare to Ship					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Conventional Processing					
Processing phase in F/H Canyons (coil and tube failure)	Once in 14,000 years	13	1.3	78,000	39
Melt and Dilute					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Melt and dilute phase (earthquake induced spill with loss of ventilation)	Once in 200,000 years	30	0.5	21,000	10
Mechanical Dilution					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Mechanical dilution phase (criticality with loss of ventilation)	Once in 33,000 years	0.71	0.074	3,000	1.5
Vitrification Technologies					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Vitrification phase (earthquake-induced release with loss of ventilation)	Once in 200,000 years	0.10	0.0017	71	0.035
Electrometallurgical Treatment					
Dry Storage phase (earthquake-induced criticality)	Once in 2,000 years	13	0.22	12,000	6.2
Electrometallurgical phase (metal melter earthquake induced spill with loss of ventilation)	Once in 200,000 years	30	0.5	21,000	10

MEI = Maximally Exposed Individual.
RBOF = Receiving Basin for Offsite Fuels.
a. All alternatives would use RBOF and the L-Reactor Disassembly Basin; therefore, accidents in these facilities are possible for each technology.
b. Not available.

considered specifically to reduce the potential for accidents with adverse consequences. During that process, assumptions about the design and operation of the proposed technologies have changed. Changes in the assumptions have resulted in changes in the outcome of the accident analyses. Details concerning the analyses are found in Appendix D of this EIS.

TC

For all of the accidents, there is a potential for injury or death to involved workers in the vicinity of the accident. In some cases, the impacts to the involved worker would be greater than to the noninvolved worker. However, prediction of latent potential health effects becomes increasingly difficult to quantify as the distance between the accident location and the receptor decreases because the individual worker exposure cannot be precisely defined with respect to the presence of shielding and other protective features. The worker also may be acutely.injured or killed by physical effects of the accident itself. DOE identified potential accidents through a detailed hazard assessment and estimated impacts using the AXAIRQ computer model (Simpkins 1995a,b), as discussed in Appendix D.

Results of accident calculations listed in Table 4.2-1 have been updated since the Draft EIS to incorporate evolution of the technology alternatives and to incorporate information that was not available at the time the Draft EIS was prepared.

TC

4.3 Construction Impacts

This section describes environmental impacts that could result from construction activities associated with SNF management at SRS. These activities would include the construction of a Transfer and Storage Facility under theNew Packaging Technology or the construction of a Transfer, Storage, and Treatment Facility under the New Processing Technology or Conventional Processing. DOE does not expect such construction activities to have appreciable impacts on geologic resources, groundwater, traffic, transportation, or cultural resources, as explained below

4.3.1 GEOLOGY AND GROUNDWATER

DOE would confine the construction of new facilities to previously disturbed and developed areas and, therefore, expects little or no environmental impacts to the geologic resources of the area. Neither the construction nor the operation of the proposed Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would affect groundwater in the area. The proposed DOE action to remove stored fuels from existing basins would eliminate a potential source of environmental releases (leaks from wet basins). The Transfer and Storage Facility or Transfer, Storage, and Treatment Facility could include the capability to perform wet receipt and unloading of SNF.

4.3.2 TRAFFIC AND TRANSPORTATION

DOE would transport construction materials, wastes, and excavated materials associated with building the proposed facilities both on and off SRS. These activities would result in increases in the operation of personal vehicles by construction workers, commercial truck traffic, and traffic associated with the daily operations of SRS. However, increases in worker and materials traffic would be small in comparison to existing traffic loads. Increased traffic congestion would be minimal.

4.3.3 CULTURAL RESOURCES

As discussed in Section 3.6, activities associated with the proposed action and alternatives for SNF management at SRS that could affect cultural resources would be the use of the three candidate sites for the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility. These sites are in reactor areas (L, C, and P) within 100 to 400 yards (91 to 366 meters) of the reactor buildings. The Savannah River Archaeological Research Program has not examined these sites. The Site Use Program, which requires a permit for clearing land on the SRS, usually initiates archaeological investigations. DOE would direct an investigation of the selected site before starting facility design and construc-

tion. Although there were homesites at or near the proposed facility sites in C and L Areas, the likelihood of historic resources surviving the construction of the reactors in the early 1950s, before the enactment of regulations to protect such resources would be small (Sassaman 1997).

The potential for the presence of prehistoric sites in the candidate locations also is limited. The L-Area site is in archaeological site density Zone 3, which has the least potential for prehistoric sites of significance. The C-Area site is in Zones 2 and 3 and has more potential. Zone 2 includes areas of moderate archaeological site density. The P-Area site is in Zone 2. However, as with any historic sites, reactor construction activities probably destroyed or severely damaged prehistoric deposits. DOE would direct an examination of the selected location for prehistoric resources before starting the design and construction of the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility (Sassaman 1997).

4.3.4 SURFACE WATER RESOURCES

Construction at SRS must comply with the requirements of South Carolina stormwater management and sediment reduction regulations, which became effective in 1992 as part of the Clean Water Act. These regulations and their associated permits require DOE to prepare erosion and sediment control plans for all projects, regardless of the land area. Runoff from the construction site would be part of a stormwater management and sedimentation control plan to minimize potential discharges of silts, solids, and other contaminants to surface-water streams. Effective January 2, 1997, the South Carolina Department of Health and Environmental Control (SCDHEC) approved General Permit coverage for stormwater management and sediment reduction at the SRS (SCDHEC 1996). Although the General Permit does not exempt any land-disturbing and construction activities from the requirements of State stormwater management and sediment control regulations, it does preclude the necessity of SCDHEC plan review and approval for land disturbing and construction activities at the SRS.

Before beginning construction, DOE would develop erosion and sediment control plans for the planned facilities. After construction and depending on the location of the construction site, the *SRS Stormwater Pollution Prevention Plan* (WSRC 1993), which is a requirement of the general NPDES stormwater permit covering industrial activities (Permit SCR000000), would include applicable erosion and sediment control measures; inclusion in the plan would not be necessary if the facility to be constructed was in the drainage area of a stormwater collection system permitted as part of NPDES Permit SC0000175.

4.3.5 AIR RESOURCES

The potential construction of facilities for the management of SNF would cause emissions of fugitive dust (particulate matter) from land-clearing activities and exhaust emissions from construction equipment (earth-moving vehicles, diesel generators). DOE has considered such impacts for activities at SRS that were similar in facility size and application and concluded that impacts to air quality would be minimal (DOE 1995a,b) and would have no effect on SRS compliance with state and Federal ambient air quality standards. Concentrations of pollutants emitted during construction activities would be at least an order of magnitude less than the South Carolina ambient air quality standards.

4.3.6 ECOLOGICAL RESOURCES

DOE is considering three brown field sites for the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility, if they are not constructed in a renovated reactor: C Area, L Area, and P Area. As noted in Section 3.4, the sites would encompass approximately 60,700 square meters (15 acres), including the main building and land required for ancillary facilities. The Treatment Facility could also be constructed on a previously disturbed site inside the F-Area or H-Area fences.

All construction activity for the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would take place within the

boundary of one of the three reactor areas in an already-developed brownfield area. Undeveloped portions of the three proposed sites provide some low-quality wildlife habitat.

Construction of the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility would involve the movement of workers and construction equipment, and would be associated with relatively loud noises from earth-moving equipment, portable generators, pile-driving equipment, pneumatic tools, drills, hammers, and the like. Although noise levels in construction areas could be as high as 110 dBA, these high local noise levels would not extend far beyond the boundaries of the project site.

Table 4.3-1 gives the attenuation of construction noise over relatively short distances. At 120 meters (400 feet) from the construction site, construction noises would range from approximately 60 to 80 dBA. Golden et al. (1980) suggest that noise levels higher than 80 to 85 dBA are sufficient to startle or frighten birds and small mammals. Thus, there would be minimal Potential for disturbing birds and small mammals outside a 120-meter radius from the construction site.

Although noise levels would be relatively low outside the immediate area of construction, the combination of construction noise and human activity probably would displace small numbers of animals (e.g., songbirds and small mammals) that could forage, feed, nest, rest, or den in the area. Construction-related disturbances are likely to create impacts to wildlife that would be small, temporary (approximately 24 months), and localized. Some animals could be driven from the area permanently, while others could become accustomed to the increased noise and activity and return to the area. Species likely to be affected (e.g., gray squirrel, opossum, white-tailed deer) are common to ubiquitous in these areas. Construction would not disturb any threatened or endangered species, would not degrade any critical or sensitive habitat, and would not affect any jurisdictional wetlands.

Table 4.3-1. Peak and attenuated noise (in dBA) levels expected from operation of construction equipment.[a]

Source	Noise level (peak)	Distance from source			
		50 feet[b]	100 feet	200 feet	400 feet
Heavy trucks	95	84-89	78-83	72-77	66-71
Dump trucks	108	88	82	76	70
Concrete mixer	105	85	79	73	67
Jackhammer	108	88	82	76	70
Scraper	93	80-89	74-82	68-77	60-71
Dozer	107	87-102	81-96	75-90	69-84
Generator	96	76	70	64	58
Crane	104	75-88	69-82	63-76	55-70
Loader	104	73-86	67-80	61-74	55-68
Grader	108	88-91	82-85	76-79	70-73
Dragline	105	85	79	73	67
Pile driver	105	95	89	83	77
Fork lift	100	95	89	83	77

a. Source: Golden et al. (1980).
b. To convert feet to meters, multiply by 0.3048.

4.3.7 IMPACTS FROM RENOVATING AN EXISTING FACILITY

4.3.7.1 Waste Generation

As discussed in Section 2.3.2.3, DOE could locate the Transfer, Storage, and Treatment Facility in a renovated reactor area, such as the 105-L facility. This would require decontamination and removal of components and systems and subsequent construction activities inside the reactor building and would result in impacts that would not occur during the construction of a virgin facility. Impacts would include generation of radioactive waste during decontamination, removal and construction. DOE has estimated that decontamination and removal and construction activities would result in the generation of approximately 476 m^3 of low-level waste over the total duration of the activities (WSRC 1998). Eventual decontamination and decommissioning (D&D) of the Transfer, Storage, and Treatment Facility (either stand-alone or in a renovated reactor facility) also would result in generation of radioactive waste.

4.3.7.2 Worker Health

DOE could locate the Transfer, Storage, and Treatment Facility in a renovated reactor area, such as the 105-L facility. This would require decontamination and removal of components and systems and subsequent construction activities inside the reactor building and would result in impacts that would not occur during the construction of a virgin facility. Impacts would include radiation exposure of workers performing these activities. The decontamination and removal and construction activities would result in a total collective worker radiation dose of 32 person-rem, based on 54 total workers and a duration of 1 year to complete all activities (Nathen 1998). The collective worker dose is estimated to result in 1.3×10^{-3} latent cancer fatalities. Eventual decontamination and decommissioning (D&D) of the Transfer, Storage, and Treatment Facility (either stand-alone or in a renovated reactor facility) also would result in radiation exposure of D&D workers.

4.3.8 SOCIOECONOMIC IMPACTS

The implementation of the alternatives discussed in this EIS could result in the construction and operation of a Transfer and Storage Facility or a Transfer, Storage and Treatment Facility, which could in turn cause incremental socioeconomic impacts in the SRS area Section 2.3.2 discusses the construction and operation of the Transfer and Storage Facility. Its construction would cost an estimated $200 million. A 2-year construction period would result in a short-term increase of fewer than 500 jobs in the region, approximately 75 percent of which would be in construction. This would be an increase in consruction jobs of approximately 2 percent (from about 16,000) and an increase of considerably less than 1 percent in total employment for the region (REMI 1995). After the 2-year period, employment would return back to its previous equilibrium. The small temporary increases in employment would not present significant impacts to the regional economy, services, or infrastructure. TC

DOE would construct the treatment phase of the Transfer, Storage, and Treatment Facility after the Transfer and Storage phase was constructed; the construction periods would not overlap. The treatment phase would require less effort to construct and would employ fewer construction employees.

None of these construction activities would significantly increase regional employment or population, and socioeconomic impacts would be negligible.

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Table 4.1-27. Fuel group and technology combination that compose the No-Action Alternative.

Fuel group	1 Prepare for Direct Co-Disposal	2 Repackage and Prepare to Ship	3 Melt and Dilute	4 Mechanical Dilution	5 Vitrification Technologies	6 Electro-metallurgical Treatment	7 Conventional Processing	8 Continued Wet Storage
A. Uranium and Thorium Metal Fuels	–	–	–	–	–	–	–	Yes
B. Materials Test Reactor-like Fuels	–	–	–	–	–	–	–	Yes
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	–	–	–	–	–	–	–	Yes
D. Loose Uranium Oxide in Cans	–	–	–	–	–	–	–	Yes
E. Higher Actinide Targets	–	–	–	–	–	–	–	Yes
F. Non-Aluminum-Clad Fuels[a]	–	–	–	–	–	–	–	Yes

a. The environmental impacts of this case were analyzed in the Programmatic SNF EIS (DOE 1995b).
HEU = highly enriched uranium.
LEU = low enriched uranium.

Table 4.1-28. Fuel group and technology combination that compose the Minimum Impact Alternative.

Fuel group	1 Prepare for Direct Co-Disposal	2 Repackage and Prepare to Ship	3 Melt and Dilute	4 Mechanical Dilution	5 Vitrification Technologies	6 Electro-metallurgical Treatment	7 Conventional Processing	8 Continued Wet Storage
A. Uranium and Thorium Metal Fuels	Yes	–	–	–	–	–	–	–
B. Materials Test Reactor-like Fuels	Yes	–	–	–	–	–	–	–
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	Yes	–	–	–	–	–	–	–
D. Loose Uranium Oxide in Cans	–	–	Yes	–	–	–	–	–
E. Higher Actinide Targets	–	Yes	–	–	–	–	–	–
F. Non-Aluminum-Clad Fuels[a]	–	Yes	–	–	–	–	–	–

a. The environmental impacts of this case were analyzed in the Programmatic SNF EIS (DOE 1995b).
HEU = highly enriched uranium.
LEU = low enriched uranium.

Table 4.1-29. Fuel group and technology combination that compose the Direct Disposal Alternative.

Fuel group	1 Prepare for Direct Co-Disposal	2 Repackage and Prepare to Ship	3 Melt and Dilute	4 Mechanical Dilution	5 Vitrification Technologies	6 Electro-metallurgical Treatment	7 Conventional Processing	8 Continued Wet Storage
A. Uranium and Thorium Metal Fuels	–	–	–	–	–	–	Yes	–
B. Materials Test Reactor-like Fuels	Yes	–	–	–	–	–	–	–
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	Yes	–	–	–	–	–	Yes[a]	–
D. Loose Uranium Oxide in Cans	–	–	Yes	–	–	–	Yes[b]	–
E. Higher Actinide Targets	–	Yes	–	–	–	–	–	–
F. Non-Aluminum-Clad Fuels[a]	–	Yes	–	–	–	–	–	–

a. For failed or sectioned Oak Ridge Reactor fuel, High-Flux Isotope Reactor fuel, and Tower Shielding Reactor fuel, Heavy Water Components Reactor fuel, and Mark-42 targets.
b. For Sterling Forest Oxide fuel.
c. The environmental impacts of this case were analyzed in the Programmatic SNF EIS (DOE 1995b).
HEU = highly enriched uranium.
LEU = low enriched uranium.

Table 4.1-30. Fuel group and technology combination that compose the Preferred Alternative.

Fuel group	1 Prepare for Direct Co-Disposal	2 Repackage and Prepare to Ship	3 Melt and Dilute	4 Mechanical Dilution	5 Vitrification Technologies	6 Electro-metallurgical Treatment	7 Conventional Processing	8 Continued Wet Storage
A. Uranium and Thorium Metal Fuels	–	–	–	–	–	–	Yes	–
B. Materials Test Reactor-like Fuels	–	–	Yes	–	–	–	–	–
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	–	–	Yes	–	–	–	Yes[a]	–
D. Loose Uranium Oxide in Cans	–	–	Yes	–	–	–	Yes[b]	–
TC E. Higher Actinide Targets	–	–	–	–	–	–	–	Yes[c]
F. Non-Aluminum-Clad Fuels[c]	–	Yes	–	–	–	–	–	–

a. For failed or sectioned Oak Ridge Reactor fuel, High-Flux Isotope Reactor fuel, and Tower Shielding Reactor fuel, Heavy Water Components Test Reactor fuel, and Mark-42 targets.
b. For Sterling Forest Oxide fuel.
TC c. The environmental impacts of this case were analyzed in the Programmatic SNF EIS (DOE 1995b).
HEU = highly enriched uranium.
LEU = low enriched uranium.
NA = not applicable; not decided in this EIS.

Table 4.1-31. Fuel group and technology combination that compose the Maximum Impact Alternative.

Fuel group	1 Prepare for Direct Co-Disposal	2 Repackage and Prepare to Ship	3 Melt and Dilute	4 Mechanical Dilution	5 Vitrification Technologies	6 Electro-metallurgical Treatment	7 Conventional Processing	8 Continued Wet Storage
A. Uranium and Thorium Metal Fuels	–	–	–	–	–	–	Yes	–
B. Materials Test Reactor-like Fuels	–	–	–	–	–	–	Yes	–
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	–	–	–	–	–	–	Yes	–
D. Loose Uranium Oxide in Cans	–	–	–	–	–	–	Yes	–
TC E. Higher Actinide Targets	–	Yes	–	–	–	–	Yes[a]	–
F. Non-Aluminum-Clad Fuels[b]	–	Yes	–	–	–	–	–	–

TC a. The environmental impacts of processing Mark-18 targets was analyzed in the Interim Management of Nuclear Materials Final Environmental Impact Statement (DOE 1995a).

b. The environmental impacts of this case were analyzed in the Programmatic SNF EIS (DOE 1995b).

HEU = highly enriched uranium.

LEU = low enriched uranium.

Table 4.1-3. Estimated maximum incremental annual dose (millirem) to noninvolved worker from airborne releases.

Fuel group	Technologies							
	1	2	3	4	5	6	7	8
	Prepare for direct co-disposal	Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing[a]	Continued wet storage
A. Uranium and Thorium Metal Fuels	0[b]	NA	5.3×10^{-4}	NA	5.3×10^{-4}	5.3×10^{-4}	3.2×10^{-4}	1.8×10^{-3}[c]
B. Materials Test Reactor-Like Fuels	0[b]	NA	0.27	0.013	0.27	0.27	0.09	0.083[c]
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0[b]	NA	0.085	0.0043	0.085	0.085	0.029	0.02[c]
D. Loose Uranium Oxide in Cans	NA	NA	5.0×10^{-3}	NA	5.0×10^{-3}	5.0×10^{-3}	5.7×10^{-3}	4.7×10^{-3}[c]
E. Higher Actinide Targets	NA	0[b]	NA	NA	NA	NA	NA	6.7×10^{-4}[c]
F. Non-Aluminum-Clad Fuels	NA	0[b]	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.
HEU = Highly Enriched Uranium.
LEU = Low Enriched Uranium.

a. Annual impacts from Conventional Processing are lower because the amount of material processed annually by this technology is less than for other technologies. The annual impacts for Conventional Processing are based on operating one dissolver in a canyon. Impacts would double if the canyon was operated at full capacity (i.e., two dissolvers). Fuel processing of the entire SNF inventory would take over 20 dissolver-years using one dissolver and about 11 dissolver-years using two dissolvers. Processing all the fuel at full capacity in a new treatment facility would take about 7 years. Appendix E provides more information related to processing durations.

b. No incremental increase expected above SRS baseline radioactive emissions values reported in Chapter 3 because these options would not change the integrity of the fuel.

c. Reflects current reactor-area emissions (including two SNF wet basins).

Table 4.1-4. Estimated maximum incremental annual dose (millirem) to hypothetical maximally exposed offsite individual from airborne releases.

	Technologies							
	1	2	3	4	5	6	7	8
Fuel group	Prepare for direct co-disposal	Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing[a]	Continued wet storage
A. Uranium and Thorium Metal Fuels	0[b]	NA	6.5×10^{-5}	NA	6.5×10^{-5}	6.5×10^{-5}	3.9×10^{-5}	2.6×10^{-4}[c]
B. Materials Test Reactor-Like Fuels	0[b]	NA	0.033	0.0016	0.033	0.033	0.011	0.012[c]
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0[b]	NA	0.010	5.2×10^{-4}	0.010	0.010	3.5×10^{-3}	3.3×10^{-3}[c]
D. Loose Uranium Oxide in Cans	NA	NA	6.1×10^{-4}	NA	6.1×10^{-4}	6.1×10^{-4}	7.0×10^{-4}	6.9×10^{-4}[c]
E. Higher Actinide Targets	NA	0[b]	NA	NA	NA	NA	NA	9.9×10^{-5}[c]
F. Non-Aluminum-Clad Fuels	NA	0[b]	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.
HEU = Highly Enriched Uranium.
LEU = Low Enriched Uranium.

a. Annual impacts from Conventional Processing are lower because the amount of material processed annually by this technology is less than for other technologies. The annual impacts for Conventional Processing are based on operating one dissolver in a canyon. Impacts would double if the canyon was operated at full capacity (i.e., two dissolvers). Fuel processing of the entire SNF inventory would take over 20 dissolver-years using one dissolver and about 11 dissolver-years using two dissolvers. Processing all the fuel at full capacity in a new treatment facility would take about 7 years. Appendix E provides more information related to processing durations.

b. No incremental increase expected above SRS baseline radioactive emissions values reported in Chapter 3 because these options would not change the integrity of the fuel.

c. Reflects current reactor-area emissions (including two SNF wet basins).

Table 4.1-5. Estimated maximum incremental annual dose (person-rem) to the 620,100 person population surrounding SRS from airborne releases.

Fuel group	Technologies							
	1	2	3	4	5	6	7	8
	Prepare for direct co-disposal	Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing[a]	Continued wet storage
A. Uranium and Thorium Metal Fuels	0[b]	NA	2.4×10^{-3}	NA	2.4×10^{-3}	2.4×10^{-3}	1.4×10^{-3}	$9.5\text{x}10^{-3}$[c]
B. Materials Test Reactor-Like Fuels	0[b]	NA	1.2	0.060	1.2	1.2	0.41	0.44[c]
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	0[b]	NA	0.38	0.019	0.38	0.38	0.13	0.12[c]
D. Loose Uranium Oxide in Cans	NA	NA	0.022	NA	0.022	0.022	0.026	0.025[c]
E. Higher Actinide Targets	NA	0[b]	NA	NA	NA	NA	NA	$3.57\text{x}10^{-3}$[c]
F. Non-Aluminum-Clad Fuels	NA	0[b]	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.
HEU = Highly Enriched Uranium.
LEU = Low Enriched Uranium.

a. Annual impacts from Conventional Processing are lower because the amount of material processed annually by this technology is less than for other technologies. The annual impacts for Conventional Processing are based on operating one dissolver in a canyon. Impacts would double if the canyon was operated at full capacity (i.e., two dissolvers). Fuel processing of the entire SNF inventory would take over 20 dissolver-years using one dissolver and about 11 dissolver-years using two dissolvers. Processing all the fuel at full capacity in a new treatment facility would take about 7 years. Appendix E provides more information related to processing durations.

b. No incremental increase expected above SRS baseline radioactive emissions values reported in Chapter 3 because these options would not change the integrity of the fuel.

c. Reflects current reactor-area emissions (including two SNF wet basins).

Table 4.1-6. Radiation doses to the public and associated latent cancer fatalities for the entire period of analysis (1998-2035).[a]

Fuel Group	Parameter	Technologies 1: Prepare for direct co-disposal	2: Repackage and prepare to ship	3: Melt and dilute	4: Mechanical dilution	5: Vitrification technologies	6: Electrometallurgical treatment	7: Conventional processing	8: Continued wet storage
A. Uranium and Thorium Metal Fuels	MEI[b] dose (millirem)	0[c]	NA	6.5×10^{-5}	NA	6.5×10^{-5}	6.5×10^{-5}	7.3×10^{-5}	0.01[g]
	MEI LCF[d,e]	0[c]	NA	3.2×10^{-11}	NA	3.2×10^{-11}	3.2×10^{-11}	3.6×10^{-11}	5.0×10^{-6}[g]
	Collective population dose (person-rem)	0[c]	NA	2.4×10^{-3}	NA	2.4×10^{-3}	2.4×10^{-3}	1.6×10^{-3}	0.36[g]
	Collective population LCF[f]	0[c]	NA	1.2×10^{-6}	NA	1.2×10^{-6}	1.2×10^{-6}	8.1×10^{-7}	1.8×10^{-4}[g]
B. Materials Test Reactor-Like Fuels	MEI[b] dose (millirem)	0[c]	NA	0.17	9.0×10^{-5}	0.17	0.17	0.54	0.46[g]
	MEI LCF[d,e]	0[c]	NA	8.5×10^{-8}	4.5×10^{-9}	8.5×10^{-8}	8.5×10^{-8}	2.7×10^{-7}	2.3×10^{-4}[g]
	Collective population dose (person-rem)	0[c]	NA	6.3	0.3	6.3	6.3	7.3	16.7[g]
	Collective population LCF[f]	0[c]	NA	3.1×10^{-3}	1.7×10^{-4}	3.1×10^{-3}	3.1×10^{-3}	3.7×10^{-3}	8.3×10^{-3}[g]
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	MEI[b] dose (millirem)	0[c]	NA	0.015	7.8×10^{-4}	0.015	0.015	0.12	0.12[g]
	MEI LCF[d,e]	0[c]	NA	7.3×10^{-9}	3.9×10^{-10}	7.3×10^{-9}	7.3×10^{-9}	6.2×10^{-8}	6.2×10^{-5}[g]
	Collective population dose (person-rem)	0[c]	NA	0.54	0.029	0.54	0.54	1.3	4.5[g]
	Collective population LCF[f]	0[c]	NA	2.7×10^{-4}	1.4×10^{-5}	2.7×10^{-4}	2.7×10^{-4}	6.5×10^{-4}	2.2×10^{-3}[g]
D. Loose Uranium Oxide in Cans	MEI[b] dose (millirem)	NA	NA	6.1×10^{-4}	NA	6.1×10^{-4}	6.1×10^{-4}	7.1×10^{-3}	0.026[g]
	MEI LCF[d,e]	NA	NA	3.0×10^{-10}	NA	3.0×10^{-10}	3.0×10^{-10}	3.6×10^{-9}	1.3×10^{-5}[g]
	Collective population dose (person-rem)	NA	NA	0.022	NA	0.022	0.022	0.075	0.95[g]
	Collective population LCF[f]	NA	NA	1.1×10^{-5}	NA	1.1×10^{-5}	1.1×10^{-5}	3.8×10^{-5}	4.7×10^{-4}[g]
E. Higher Actinide Targets	MEI[b] dose (millirem)	NA	0[c]	NA	NA	NA	NA	NA	3.7×10^{-3}[g]
	MEI LCF[d,e]	NA	0[c]	NA	NA	NA	NA	NA	1.9×10^{-6}[g]
	Collective population dose (person-rem)	NA	0[c]	NA	NA	NA	NA	NA	0.14[g]
	Collective population LCF[f]	NA	0[c]	NA	NA	NA	NA	NA	6.8×10^{-5}[g]
F. Non-Aluminum-Clad Fuels	MEI[b] dose (millirem)	NA	0[c]	NA	NA	NA	NA	NA	NA
	MEI LCF[d,e]	NA	0[c]	NA	NA	NA	NA	NA	NA
	Collective population dose (person-rem)	NA	0[c]	NA	NA	NA	NA	NA	NA
	Collective population LCF[f]	NA	0[c]	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.
HEU = Highly Enriched Uranium.
LEU = Low Enriched Uranium.

a. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.
b. MEI = Maximally Exposed Individual; i.e., a hypothetical member of the public whose location and habits result in exposure to the maximum dose from all pathways.
c. No incremental increase expected above SRS baseline radioactive emissions values presented in Chapter 3 because these options would not affect the integrity of the fuel.
d. LCF = latent cancer fatalities.
e. For an individual, the LCF value should be interpreted statistically; e.g., 1×10^{-9} = 1 chance in 1 billion to develop a fatal cancer.
f. For collective population, the LCF value should be interpreted as the number of cancers that could be expected in the population.

g. Reflects current reactor-area emissions (including two SNF wet basins) for the entire period of analysis.

Table 4.1-7. Number of radiation workers and collective worker radiation dose (person-rem) and associated latent cancer fatalities for the entire period of analysis (1998-2035).[a]

		Technologies							
		1	2	3	4	5	6	7	8
Fuel Group	Parameter	Prepare for direct co-disposal	Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing	Continued wet storage
	Number of radiation workers[b]	75	38	100	88	159	119	150	40
A. Uranium and Thorium Metal Fuels	Collective worker dose (person-rem)	11	NA	12	NA	15	13	18	12
	LCF[c]	4.2×10^{-3}	NA	4.8×10^{-3}	NA	6.1×10^{-3}	5.2×10^{-3}	7.2×10^{-3}	4.9×10^{-3}
B. Materials Test Reactor-Like Fuels	Collective worker dose (person-rem)	480	NA	530	520	680	580	1,300	560
	LCF	0.19	NA	0.21	0.21	0.27	0.23	0.50	0.22
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	Collective worker dose (person-rem)	140	NA	150	150	190	160	600	150
	LCF	0.054	NA	0.059	0.059	0.075	0.064	0.24	0.060
D. Loose Uranium Oxide in Cans	Collective worker dose (person-rem)	NA	NA	31	NA	40	34	170	32
	LCF	NA	NA	0.012	NA	0.016	0.014	0.069	0.013
E. Higher Actinide Targets	Collective worker dose (person-rem)	NA	3	NA	NA	NA	NA	NA	5
	LCF	NA	1.3×10^{-3}	NA	NA	NA	NA	NA	1.8×10^{-3}
F. Non-Aluminum Clad Fuels	Collective worker dose (person-rem)	NA	26	NA	NA	NA	NA	NA	NA
	LCF	NA	0.011	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.
HEU = Highly Enriched Uranium.
LEU = Low Enriched Uranium.

a. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.

b. Estimates of the number of radiation workers are based on past operating experience (Bickford et al. 1997).

c. LCF = latent cancer fatalities.

Table 4.1-8. Radiation doses to the maximally exposed noninvolved worker (at 640 meters) and associated latent cancer fatalities for the entire period of analysis (1998-2035).[a]

			Technologies							
			1	2	3	4	5	6	7	8
	Fuel Group	Parameter	Prepare for direct co-disposal	Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing	Continued wet storage
A.	Uranium and Thorium Metal Fuels	Noninvolved worker dose (millirem)	0[c]	NA	5.3×10^{-4}	NA	5.3×10^{-4}	5.3×10^{-4}	3.2×10^{-4}	0.068[d]
		Noninvolved worker LCF[b]	0[c]	NA	2.1×10^{-10}	NA	2.1×10^{-10}	2.1×10^{-10}	1.3×10^{-10}	2.7×10^{-5}[d]
B.	Materials Test Reactor-Like Fuels	Noninvolved worker dose (millirem)	0[c]	NA	1.4	0.074	1.4	1.4	1.3	3.1[d]
		Noninvolved worker LCF[b]	0[c]	NA	5.6×10^{-7}	2.9×10^{-8}	5.6×10^{-7}	5.6×10^{-7}	5.4×10^{-7}	1.3×10^{-3}[d]
C.	HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	Noninvolved worker dose (millirem)	0[c]	NA	0.12	6.3×10^{-3}	0.12	0.12	0.22	0.84[d]
		Noninvolved worker LCF[b]	0[c]	NA	4.8×10^{-8}	2.5×10^{-9}	4.8×10^{-8}	4.8×10^{-8}	8.6×10^{-8}	3.4×10^{-4}[d]
D.	Loose Uranium Oxide in Cans	Noninvolved worker dose (millirem)	NA	NA	5.0×10^{-3}	NA	5.0×10^{-3}	5.0×10^{-3}	0.013	0.18[d]
		Noninvolved worker LCF[b]	NA	NA	2.0×10^{-9}	NA	2.0×10^{-9}	2.0×10^{-9}	5.0×10^{-9}	7.1×10^{-5}[d]
E.	Higher Actinide Targets	Noninvolved worker dose (millirem)	NA	0[c]	NA	NA	NA	NA	NA	0.025[d]
		Noninvolved worker LCF[b]	NA	0[c]	NA	NA	NA	NA	NA	1.0×10^{-5}[d]
F.	Non-Aluminum-Clad Fuels	Noninvolved worker dose (millirem)	NA	0[c]	NA	NA	NA	NA	NA	NA
		Noninvolved worker LCF[b]	NA	0[c]	NA	NA	NA	NA	NA	NA

NA = Technology is not applicable to this fuel type.
HEU = Highly Enriched Uranium.
LEU = Low Enriched Uranium.

a. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.
b. LCF = latent cancer fatalities; this number should be interpreted statistically.
c. No incremental increase expected above SRS baseline radioactive emissions values presented in Chapter 3, because these options would not affect the integrity of the fuel.
d. Reflects current reactor-area emissions (including two SNF wet basins) for the entire period of analysis.

Table 4.1-10. High-level waste generation for the entire period of analysis (1998-2035) (cubic meters).[a,b,c]

		Technologies							
		1	2	3	4	5	6	7	8
Fuel group	Parameter	Prepare for direct co-disposal	Repackage and prepare to ship	Melt and dilute	Mechanical dilution	Vitrification technologies	Electrometallurgical treatment	Conventional processing	Continued Wet storage
A. Uranium and Thorium Metal Fuels	Liquid high-level waste	10	NA	10	NA	10	10	170	36
	Equivalent DWPF canisters	<1	NA	<1	NA	<1	<1	3	<1
	Saltstone	26	NA	26	NA	26	26	430	97
B. Materials Test Reactor-Like Fuels	Liquid high-level waste	470	NA	450	470	450	450	7,700	1,700
	Equivalent DWPF canisters	8	NA	7	8	7	7	120	28
	Saltstone	1,250	NA	1,200	1,300	1,200	1,200	20,000	4,500
C. HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging	Liquid high-level waste	125	NA	120	130	120	120	2,100	450
	Equivalent DWPF canisters	2	NA	2	2	2	2	32	8
	Saltstone	330	NA	320	340	320	320	5,400	1,200
D. Loose Uranium Oxide in Cans	Liquid high-level waste	NA	NA	25	NA	25	25	450	96
	Equivalent DWPF canisters	NA	NA	<1	NA	<1	<1	7	2
	Saltstone	NA	NA	67	NA	67	67	1,100	260
E. Higher Actinide Targets	Liquid high-level waste	NA	4	NA	NA	NA	NA	NA	14
	Equivalent DWPF canisters	NA	<1	NA	NA	NA	NA	NA	<1
	Saltstone	NA	10	NA	NA	NA	NA	NA	36
F. Non- Aluminum-Clad Fuels	Liquid high-level waste	NA	30	NA	NA	NA	NA	NA	NA
	Equivalent DWPF canisters	NA	<1	NA	NA	NA	NA	NA	NA
	Saltstone	NA	80	NA	NA	NA	NA	NA	NA

DWPF = Defense Waste Processing Facility.
NA = Technology is not applicable to this fuel type.
HEU = Highly Enriched Uranium.
LEU = Low Enriched Uranium.
a. Except DWPF canisters.
b. To convert cubic meters to cubic yards, multiply by 1.308.
c. Potentially reduced fuel receipts could reduce the reported impacts. Scaling factors applied to these impact values should be applied specifically to each fuel group affected. For example, if the amount of fuel in Group B were reduced to 80 percent of the value reported in Table 1-1, then each value reported for Group B should be multiplied by 0.8.

CHAPTER 5. CUMULATIVE IMPACTS

The Council on Environmental Quality (CEQ) regulations implementing the procedural provisions of the National Environmental Policy Act (NEPA) define cumulative impacts as the impacts on the environment which result from the incremental impact of the action when added to other past, present, and reasonably foreseeable future actions regardless of what agency (Federal or non-Federal) or person undertakes such other actions (40 CFR 1508.7). The cumulative impacts analysis presented in this section is based on the incremental actions associated with the maximum impact alternative for spent nuclear fuel (SNF) management at the Savannah River Site (SRS), other actions associated with onsite activities, and offsite activities with the potential for related environmental impacts. Although it is unlikely that the maximum impact alternative would be implemented to manage SNF at SRS, it was used to estimate cumulative impacts to ensure a conservative analysis. In accordance with a handbook recently prepared by CEQ (1997), the U.S. Department of Energy (DOE) identified the resource areas in which SNF management could add to the impacts of past, present, and reasonably foreseeable actions within the project impact zones as defined by CEQ (1997).

Based on an examination of the environmental impacts of direct and indirect SNF management actions coupled with DOE and other agency actions, it was determined that cumulative impacts for the following areas need to be presented: (1) air resources; (2) water resources; (3) public and worker health; (4) waste generation; (5) utilities and energy consumption; and (6) socioeconomics. Discussion of cumulative impacts for the following resources is omitted because impacts from the proposed SNF management activities would be so small that their potential contribution to cumulative impacts would be negligible: geologic resources, ecological resources, aesthetic and scenic resources, cultural resources, and traffic. EC

For determining the impact to air, water, human health, waste generation, utilities and energy, and socioeconomic resources from commercial and Federal nuclear facilities, the 50-mile (80-kilometer) radius surrounding SRS was selected as the project impact zone. For aqueous releases, the downstream population that uses the Savannah River as its source of drinking water was included in the project impact zone.

Nuclear facilities within a 50-mile radius of SRS include Georgia Power's Plant Vogtle Electric Generating Plant across the river from SRS; Chem-Nuclear Inc., a commercial low-level waste burial site just east of SRS; and Starmet CMI, Inc. (formerly Carolina Metals), located southeast of SRS, which processes uranium-contaminated metals. Radiological impacts from the operation of the Vogtle Electric Generating Plant, a two-unit commercial nuclear power plant are minimal, but DOE has factored them into the analysis. The South Carolina Department of Health and Environmental Control Annual Report (SCDHEC 1995) indicates that operation of the Chem-Nuclear Services facility and the Starmet CMI facility do not noticeably impact radiation levels in air or liquid pathways in the vicinity of SRS. Therefore, they are not included in this assessment.

The counties surrounding SRS have numerous existing (e.g., textile mills, paper product mills, and manufacturing facilities) and planned (e.g., Bridgestone Tire) industrial facilities with permitted air emissions and discharges to surface waters. Because of the distances between SRS and the private industrial facilities, there is little opportunity for interactions of plant emissions, and no major cumulative impact on air or water quality. Construction and operation of Bridgestone Tire and Hankook Polyester facilities could affect the regional socioeconomic cumulative impacts. TC

Additional offsite facilities with the potential to affect the nonradiological environment include South Carolina Electric and Gas Company's Urquhart Station. Urquhart Station is a three-unit, 250-megawatt, coal- and natural-gas-fired steam electric plant in Beech Island, South Carolina, located about 32 river kilometers (20 river miles) north of SRS. Because of the distance between SRS and the Urquhart Station and the regional wind direction frequencies, there is little opportunity for any interaction of plant emissions, and no significant cumulative impact on air quality.

EC | DOE also evaluated the impacts from its own proposed future actions by examining impacts to resources and the human environment as shown in NEPA documentation related to SRS (see Section 1.6). Additional NEPA documents related to SRS that are considered in the cumulative impacts section include the following:

Final Environmental Impact Statement - Interim Management of Nuclear Materials (DOE/EIS-0220) (DOE 1995a). DOE has begun implementation of the preferred alternatives for the nuclear materials discussed in the Interim Management of Nuclear Materials EIS. SRS baseline data in this chapter reflect projected impacts from implementation.

EC |

TC |

Final Environmental Impact Statement for the Accelerator Production of Tritium at Savannah River Site (DOE/EIS-0270) (DOE 1999a). DOE has proposed an accelerator design (using helium-3 target blanket material) and an alternate accelerator design (using lithium-6 target blanket material). If an accelerator is built, it would be located at SRS. However, since the Record of Decision states the preferred alternative as use of an existing commercial light-water reactor, data from this EIS are not used.

EC |

TC |

Environmental Assessment for the Tritium Facility Modernization and Consolidation Project at the Savannah River Site (DOE/EA-1222) (DOE 1997). This environmental assessment (EA) addresses the impacts of consolidating the tritium activities currently the new Building 233-H and Building 234-H. Tritium extraction functions would be transferred to Tritium Extraction Facility. The overall impact would be to reduce the tritium facility complex net tritium emissions by up to 50 percent. Another positive effect of this planned action would be to reduce the amount of low-level radioactive job-control waste. Effects on other resources would be negligible. Therefore, impacts from the environmental assessment have not been included in this cumulative impacts analysis.

EC |

Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement (DOE/EIS-0240) (DOE 1996). This cumulative impacts analysis incorporates the alternative of blending at SRS highly enriched uranium to 4 percent low-enriched uranium as uranyl nitrate hexahydrate as stated in the Record of Decision (61 FR 40619, August 5, 1996).

EC |

TC |

Final Environmental Impact Statement on Management of Certain Plutonium Residues and Scrub Alloy Stored at the Rocky Flats Environmental Technology Site (DOE/EIS-0277F) (DOE 1998). DOE proposes to process certain plutonium-bearing materials being stored at the Rocky Flats Environmental Technology Site. These materials are plutonium residues and scrub alloy remaining from nuclear weapons manufacturing operations formerly conducted by DOE at Rocky Flats. DOE has decided to remove the plutonium from certain residues that would be shipped from the Rocky Flats Environmental Technology Site to SRS for stabilization. The separated plutonium would be stored at SRS pending disposition decisions. Environmental impacts from using F Canyon to chemically separate the plutonium from the remaining materials at SRS are included in this section.

EC |

EC

EC

Final Environmental Impact Statement for the Construction and Operation of a Tritium Extraction Facility at the Savannah River Site (DOE/EIS-0271) (DOE 1999b). As stated in the Record of Decision (64 FR 26369; 5/14/99), DOE will construct and operate a Tritium Extraction Facility on SRS to provide the capability to extract tritium from commercial light water reactor targets and targets of similar design. The purpose of the proposed action and alternatives evaluated in the EIS is to provide tritium extraction capability to support either accelerator or reactor production. Environmental impacts from the maximum processing option in this EIS are included in this section.

EC

TC

Surplus Plutonium Disposition Final Environmental Impact Statement (DOE/EIS-0283) (DOE 1999c). This EIS analyzes the activities necessary to implement DOE's disposition strategy for surplus plutonium. In January 2000 DOE issued a Record of Decision selecting SRS as the site for all three disposition facilities: mixed-oxide fuel fabrication, plutonium immobilization, and plutonium pit disassembly and conversion. Impacts from these facilities are included in this section.

EC

Defense Waste Processing Facility Supplemental Environmental Impact Statement (DOE/EIS-0082-S) (DOE 1994). The selected alternative in the Record of Decision (ROD) was the completion and operation of the Defense Waste Processing Facility to immobilize high-level radioactive waste at the SRS. The facility is currently processing sludge from SRS high-level waste tanks. However, SRS baseline data is not representative of full DWPF operational impacts, including processing of salt and supernate from these tanks. Therefore, the DWPF data is listed separately.

L2-16
EC

Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel (DOE/EIS-0306D) (DOE 1999d). DOE has published a draft environmental impact statement (64 FR 8553, 2/22/99) for treatment of sodium-bonded spent nuclear fuel. Two of the alternatives being evaluated in the Treatment and Management EIS are to process INEEL's sodium-bonded fuel inventory at SRS using the Plutonium-Uranium Extraction (PUREX) process and to use the Melt and Dilute facility being proposed in the EIS. Because processing at SRS is a reasonable alternative to processing at INEEL, it is being included in the Spent Nuclear Fuel Management EIS cumulative impact analysis. These methods could be used for the sodium-bonded spent nuclear fuel blanket assemblies currently in storage at INEEL. There are approximately 22.4 MTHM of Experimental Breeder Reactor-II (EBR-II) fuel blankets and 34.2 MTHM of Fermi-1 fuel blankets to be processed. This fuel would be declad before shipment to SRS. Because the decladding activities would occur at INEEL, the impacts of these decladding activities are not included in this chapter.

L2-16
TC

This EIS includes cumulative impacts of sodium-bonded spent nuclear fuel processing at the SRS based on data from the Draft Electrometallurgical Treatment EIS. Data used in this EIS are based on Purex processing at SRS, which is more is conservative.

DOE is currently evaluating nuclear material disposition needs. Other material discussed for processing at SRS under the PNA include single-pass reactor SNF at Hanford, a small amount of damaged SNF at Idaho National Engineering and Environmental Laboratory (INEEL), classified fissile material metal parts at the Rocky Flats Environmental Technology Site (RFETS), and plutonium scrap at Hanford. Currently, DOE has no plan or proposal to transfer the single-pass reactor SNF at Hanford or the damaged SNF at INEEL to SRS so that material was not considered for the cumulative impacts under this EIS. In an amended Record of Decision for the *Final Environmental Impact Statement on Storage and Disposition of Surplus Fissile Material,*

DOE decided to transfer classified metal from RFETS to SRS for stabilization and storage. DOE is considering transferring the plutonium scrap from Hanford to SRS for stabilization and storage pending appropriate National Environmental Policy Act review. As a result, DOE has included processing that material as part of the cumulative impacts for this EIS.

L2-16 DOE is continuing to evaluate the inventory of nuclear material at facilities throughout the DOE complex. DOE's Nuclear Material Integration initiative is one such recent effort that has identified material which could be processed at SRS. Although there are no current plans to process these materials at SRS, DOE considers it appropriate to include a qualitative estimate of impacts as part of the cumulative impacts for this EIS because it is not unforeseen that processing at SRS could occur.

EC In addition, the cumulative impacts analysis includes the impacts from actions proposed in this SNF EIS. Risks to members of the public and site workers from radiological and nonradiological releases are based on operational impacts from the maximum impact alternative described in Section 4.1.2.

In addition, the cumulative impacts analysis accounts for other SRS operations. Most of the
EC SRS baseline data are based on 1997 environmental report information (Arnett and Mamatey
EC 1998), which are the most recent published data available.

Temporal limits were defined by examining the period of influence from both the proposed action and other Federal and non-Federal actions that have the potential for cumulative impacts. Actions for SNF management are expected to begin
TC in 2000 in preparation for ultimate offsite disposal, possibly in a monitored geologic repository which probably will not be available until at least 2010. Final offsite shipments of SNF from SRS for disposal would be completed by 2035.

The period of interest for the cumulative impacts analysis for this SNF EIS includes the potential construction and operation of the Tritium Extraction Facility and while actions for management of nuclear materials, highly enriched uranium, surplus plutonium disposition, and sodium-bonded nuclear fuel would be ongoing. TC

5.1 Air Resources

Table 5-1 compares the cumulative concentrations of nonradiological air pollutants from the SRS to Federal and state regulatory standards. The listed values are the maximum modeled concentrations that could occur at ground level at the Site boundary. The data demonstrate that total estimated concentrations of nonradiological air pollutants from SRS would in all cases be below the regulatory standards at the Site boundary. The highest percentages of the regulatory standards are for sulfur dioxide concentrations for the shorter time interval (approximately 97 percent of standard for the 24-hour averaging time), for particulate matter of less than 10 microns (approximately 89 per- cent of standard for the 24-hour averaging time), and total suspended particulates (approximately 90 percent of standard on an annual basis). The remaining pollutant emissions would range from 1 to 69 percent of the applicable standards. TC TC TC

The majority of the impacts come from estimates of SRS baseline concentrations. It is unlikely that actual concentrations at ambient monitoring stations would be as high as that shown for the baseline values. The SRS baseline values are based on maximum potential emissions from the 1998 air emissions inventory and for all SRS sources, and observed concentrations from nearby ambient air monitoring stations. TC

Table 5-1. Estimated maximum cumulative ground-level concentrations of nonradiological pollutants (micrograms per cubic meter) at SRS boundary.[a,b]

Pollutant	Averaging time	SCDHEC ambient standard ($\mu g/m^3$)	SNF	SRS baseline ($\mu g/m^3$)	Other foreseeable planned SRS activities[c] ($\mu g/m^3$)	Cumulative concentration[d,e] ($\mu g/m^3$)	Percent of standard
Carbon monoxide	1 hour	40,000	9.760	10,000	36.63	10,046	25
	8 hours	10,000	1.31	6,900	5.15	6,906	69
Oxides of Nitrogen	Annual	100	3.36	26	4.38	33.7	34
Sulfur dioxide	3 hours	1,300	0.98	1,200	8.71	1,210	93
	24 hours	365	0.13	350	2.48	352.6	97
	Annual	80	0.02	34	0.17	34.2	43
Ozone[f]	1 hour	235	0.80	NA[g]	0.71	1.5	1
Lead	Max. quarter	1.5	NA	0.03	0.00	0.03	2
Particulate matter (≤10 microns aerodynamic diameter)[f]	24 hours	150	0.13	130	3.24	133.4	89
	Annual	50	0.02	25	0.13	25.2	50
Total suspended particulates ($\mu g/m^3$)	Annual	75	0.02	67	0.06	67.1	89

L4-17 TC

a. DOE (1994; 1996; 1998; 1999b,c,d) and Hunter (1999) for baseline values.
b. Hydrochloric acid, formaldehyde, hexane, and nickel are not listed in Table 5-1 because operation of SNF or other foreseeable, planned SRS activities would not result in any change to the SRS baseline concentrations of these toxic pollutants.
c. Includes Highly Enriched Uranium, Tritium Extraction Facility, Management of Certain Plutonium Residues and Scrub Alloy Concentrations, Defense Waste Processing Facility, and Disposition of Surplus Plutonium, Sodium-Bonded Spent Nuclear Fuel, and components from throughout the DOE complex.
d. SCDHEC (1976).
e. Includes SNF concentrations.
f. New NAAQS for ozone (1 hr replaced by 8 hr standard = 0.08 ppm) and particulate matter ≤ 2.5 microns (24 hr standard = 65 $\mu g/m^3$) and annual standard of 15 $\mu g/m^3$ will become enforceable during the stated temporal range of the cumulative impacts analyses.
g. Not available.

TC

DOE also evaluated the cumulative impacts of airborne radioactive releases in terms of dose to a maximally exposed individual at the SRS boundary. DOE included the impacts of Plant Vogtle (NRC 1996) in this cumulative total. The radiological emissions from the operation of the Chem-Nuclear low-level waste disposal facility just east of SRS are very low (SCDHEC 1992) and are not included.

Table 5-2 lists the results of this analysis, using 1997 emissions (1992 for Plant Vogtle) for the SRS baseline. The cumulative dose to the maximally exposed member of the public would be 1 x 10^{-4} rem (or 0.1 millirem) per year, well below the regulatory standard of 10 millirem per year (40 CFR Part 61). Summing the doses to maximally exposed individual for the nine actions and baseline SRS operations listed in Table 5-2 is an extremely conservative approach because in order to get the calculated dose, the maximally exposed individual would have to occupy different physical locations at the same time, which is impossible.

TC TC

Table 5-2. Estimated average annual cumulative radiological doses and resulting health effects to the maximally exposed offsite individual and population in the 50-mile radius from airborne releases.

Activity	Offsite Population			
	Maximally exposed individual		50-mile population	
	Dose (rem)	Probability of fatal cancer risk	Collective dose (person-rem)	Excess latent cancer fatalities
SRS Baseline[a]	5.0×10^{-5}	2.5×10^{-8}	2.2	1.1×10^{-3}
Management of Spent Nuclear Fuel[b]	1.5×10^{-5}	7.5×10^{-9}	0.56	2.8×10^{-4}
Surplus HEU Disposition[c]	2.5×10^{-6}	1.3×10^{-9}	0.16	8.0×10^{-5}
Tritium Extraction Facility[d]	2.0×10^{-5}	1.0×10^{-8}	0.77	3.9×10^{-4}
Surplus Plutonium Disposition[e]	7.4×10^{-6}	3.7×10^{-9}	1.8	9.0×10^{-4}
Management of Plutonium Residues/ Scrub Alloy[f]	5.7×10^{-7}	2.9×10^{-10}	$6.2\text{x}10^{-3}$	3.1×10^{-6}
Defense Waste Processing Facility[g]	1.0×10^{-6}	5.0×10^{-10}	0.071	3.6×10^{-5}
DOE complex miscellaneous components[h]	4.4×10^{-6}	2.2×10^{-9}	7.0×10^{-3}	3.5×10^{-6}
Sodium-Bonded Spent Nuclear Fuel[i]	3.9×10^{-7}	2.0×10^{-10}	1.9×10^{-2}	9.5×10^{-6}
Plant Vogtle[j]	5.4×10^{-7}	2.7×10^{-10}	0.042	$2.1\text{x}10^{-5}$
Total	1.0×10^{-4}	5.1×10^{-8}	5.6	2.8×10^{-3}

a. Arnett and Mamatey (1998) for 1997 data for MEI and population.
b. Maximum-impact alternative.
c. DOE (1996); HEU = highly enriched uranium.
d. DOE (1999b).
e. DOE (1999c).
f. DOE (1998).
g. DOE (1994).
h. Derive from impacts from conventional processing of Group A fuel.
i. DOE (1999d).
j. NRC (1996).

Adding the population doses from current and projected activities at SRS, Plant Vogtle, and management of SNF could yield a total annual cumulative dose of 5.6 person-rem from airborne sources. The total annual cumulative dose translates into $2.8\text{x}10^{-3}$ latent cancer fatality for each year of exposure for the population living within a 50-mile (80-kilometer) radius of the SRS. For comparison, 143,863 deaths from cancer due to all causes would be likely in the same population over their lifetimes.

5.2 Water Resources

At present, a number of SRS facilities discharge treated wastewater to Upper Three Runs and its tributaries and Fourmile Branch via National Pollutant Discharge Elimination System (NPDES)–permitted outfalls. These include the F and H Area Effluent Treatment Facility (ETF) and the M-Area Liquid Effluent Treatment Facility. As stated in Section 4.1.1.1, SNF operations are not expected to result in any discharges to groundwater. The only technology that would result in discharges of radioactive and nonradioactive effluents to surface water would be Conventional Processing. The major sources of liquid effluents from facilities associated with Conventional Processing would be process cooling water and steam condensate systems that could contain small quantities of radionuclides and chemicals. This process wastewater would be treated at ETF and then discharged to Upper Three Runs. Studies of water quality and biota

downstream of the ETF outfall suggest that discharges from it have not degraded the water quality of Upper Three Runs. Other potential sources of contaminants into Upper Three Runs during the SNF management period include the accelerator production of tritium, the tritium extraction facility, environmental restoration, and decontamination and decommissioning activities, as well as modifications to existing SRS facilities. Discharges associated with the accelerator production of tritium and tritium extraction facility activities would not add significant amounts of nonradiological contaminants to Upper three Runs. The amount of discharge associated with environmental restoration and decontamination and decommissioning activities would vary based on the level of activity. All the potential activities that could result in wastewater discharges would be required to comply with the NPDES permit limits that ensure protection of water quality. Studies of water quality and biota in Upper Three Runs suggest that discharges from facilities outfalls have not degraded the stream (Halverson et al. 1997).

Table 5-3 summarizes the estimated cumulative radiological doses from waterborne sources to human receptors downstream from SRS. Liquid effluents would be released to SRS streams that are tributaries of the Savannah River could contain small quantities of radionuclides. The exposure pathways considered in this analysis included drinking water, fish ingestion, shoreline exposure, swimming, and boating. The estimated cumulative dose to the maximally exposed member of the public from liquid releases would be 2.4×10^{-4} rem (or 0.24 millirem) per year, well below the regulatory standard of 4 millirem per year (40 CFR Part 141). Adding the population doses associated with current and projected SRS activities would yield a cumulative annual dose of 2.6 person-rem from liquid sources. This translates into 0.0013 latent cancer fatality for each year of exposure of the population living within a 50-mile (80-kilometer) radius of the SRS. For comparison, 15,300 deaths from cancer due to all causes would be likely in the population of 70,000 downstream residents over their lifetimes.

5.3 Public and Worker Health

Table 5-4 summarizes the cumulative radiological health effects of routine SRS operations, proposed DOE actions, and non-Federal nuclear facility operations (Plant Vogtle Electric Generating Facility). Impacts resulting from proposed DOE actions are described in the EISs listed previously in this chapter. In addition to estimated radiological doses to the hypothetical maximally exposed offsite individual, the offsite population, and involved workers, Table 5-4 also lists the potential number of latent cancer fatalities for the public and workers due to exposure to radiation. The radiation dose to the maximally exposed offsite individual from air and liquid pathways would be 3.4×10^{-4} rem (0.34 mrem) per year, which is well below the applicable DOE regulatory limits (10 mrem per year from the air pathway, 4 mrem per year from the liquid pathway, and 100 mrem per year for all pathways). The total annual population dose for current and projected activities of 8.2 person-rem translates into 0.004 latent cancer fatality for each year of exposure for the population living within a 50-mile (80-kilometer) radius of the SRS. As stated in Section 5.1, for comparison, 143,863 deaths from cancer due to all causes would be likely in the same population over their lifetimes.

The annual radiation dose to the involved worker population would be 859 person-rem. In addition, doses to individual workers would be kept below the regulatory limit of 5,000 mrem per year (10 CFR 835). Furthermore, as low as reasonably achievable principles would be exercised to maintain individual worker doses below the DOE Administrative Control Level of 2,000 mrem per year.

Table 5-3. Estimated average annual cumulative radiological doses and resulting health effects to offsite population in the 50-mile radius from aqueous releases.

	Offsite Population			
	Maximally exposed individual		50-mile population	
Activity	Dose (rem)	Probability of fatal cancer risk	Collective dose (person-rem)	Excess latent cancer fatalities
SRS Baseline[a]	1.3×10^{-4}	6.5×10^{-8}	2.4	1.1×10^{-3}
Management of Spent Nuclear Fuel[b]	5.7×10^{-5}	2.9×10^{-8}	0.19	9.5×10^{-5}
Surplus HEU Disposition[c]	(d)	(d)	(d)	(d)
Tritium Extraction Facility[e]	(d)	(d)	(d)	(d)
Defense Waste Processing Facility[f]	(d)	(d)	(d)	(d)
Surplus Plutonium Disposition[g]	(d)	(d)	(d)	(d)
Management Plutonium Residues/Scrub Alloy[h]	(d)	(d)	(d)	(d)
DOE complex miscellaneous components[i]	4.2×10^{-8}	2.1×10^{-11}	2.4×10^{-4}	1.2×10^{-7}
Sodium-Bonded Spent Nuclear Fuel[j]	1.2×10^{-7}	6.0×10^{-11}	6.8×10^{-4}	3.4×10^{-7}
Plant Vogtle[k]	5.4×10^{-5}	2.7×10^{-8}	2.5×10^{-3}	1.3×10^{-6}
Total	2.4×10^{-4}	1.2×10^{-7}	2.6	1.3×10^{-3}

a. Arnett and Mamatey (1998) for 1997 data for MEI and population. Worker dose is based on 1997 data (WSRC 1998).
b. Maximum-impact alternative.
c. DOE (1996); HEU = highly enriched uranium.
d. Less than minimum reportable levels.
e. DOE (1999b).
f. DOE (1994).
g. DOE (1999c).
h. DOE (1998).
i. Derived from impacts from conventional processing.
j. DOE (1999d).
k. NRC (1996).

5.4 Waste Generation

As stated in Section 4.1.1.4, high-level waste, transuranic waste, and low-level waste would be generated from SNF management activities. Smaller amounts of mixed and hazardous waste would also be generated from SNF processing activities. The largest volume of high-level and transuranic waste would be generated with the Conventional Processing alternative. However, as stated in Section 4.1.1.4, the projected high-level waste and transuranic waste generation rates would not require additional treatment and storage capacities beyond the current and planned SRS capacities. In general, the waste generation rate varies with each phase of SNF handling and the type of fuel group. The total radioactive/hazardous waste volume associated with SNF activities could range from 20,700 cubic meters (27,076 cubic yards) for the minimum impact option to 154,967 cubic meters (202,681 cubic yards) for the maximum impact (conventional processing) option.

Table 5-4. Estimated average annual cumulative radiological doses and resulting health effects to offsite population and facility workers.

	Maximally exposed individual				Offsite population[a]				Workers	
Activity	Dose from airborne releases (rem)	Dose from liquid releases (rem)	Total dose (rem)	Probability of fatal cancer risk	Collective dose from airborne releases (person-rem)	Collective dose from liquid releases (person-rem)	Total collective dose (person-rem)	Excess latent cancer fatalities	Collective dose	Excess latent cancer fatalities
SRS Baseline[b]	5.0×10^{-5}	1.3×10^{-4}	1.8×10^{-4}	9.0×10^{-8}	2.2	2.4	4.6	2.3×10^{-3}	165	0.066
Management of Spent Nuclear Fuel[c]	1.5×10^{-5}	5.7×10^{-5}	7.2×10^{-5}	3.6×10^{-8}	0.56	0.19	0.75	3.8×10^{-4}	55	0.022
Surplus HEU Disposition[d]	2.5×10^{-6}	(e)	2.5×10^{-6}	1.3×10^{-8}	0.16	(e)	0.16	8.0×10^{-5}	11	4.4×10^{-3}
Tritium Extraction Facility[f]	2.0×10^{-5}	(e)	2.0×10^{-5}	1.0×10^{-8}	0.77	(e)	0.77	3.9×10^{-4}	4	1.6×10^{-3}
Defense Waste Processing Facility[g]	1.0×10^{-6}	(e)	1.0×10^{-6}	5.0×10^{-10}	0.071	(e)	0.071	3.6×10^{-5}	120	0.048
Surplus Plutonium Disposition[h]	7.4×10^{-6}	(e)	7.4×10^{-6}	3.7×10^{-9}	1.8	(e)	1.8	9.0×10^{-4}	456	0.18
Management Plutonium Residues/ Scrub Alloy[i]	5.7×10^{-7}	(e)	5.7×10^{-7}	2.9×10^{-10}	6.2×10^{-3}	(e)	6.2×10^{-3}	3.1×10^{-6}	7.6	3×10^{-3}
DOE complex miscellaneous components[j]	4.4×10^{-6}	4.2×10^{-8}	4.4×10^{-6}	2.2×10^{-9}	7.0×10^{-3}	2.4×10^{-4}	7.2×10^{-3}	3.6×10^{-6}	2	0.001
Sodium-Bonded Spent Nuclear Fuel[k]	3.9×10^{-7}	1.2×10^{-7}	5.1×10^{-7}	2.6×10^{-10}	1.9×10^{-2}	6.8×10^{-4}	2.0×10^{-2}	9.8×10^{-6}	38	0.015
Plant Vogtle[l]	5.4×10^{-7}	5.4×10^{-5}	5.5×10^{-5}	2.7×10^{-8}	0.042	2.5×10^{-3}	0.045	2.2×10^{-5}	NA	NA
Total	1.0×10^{-4}	2.4×10^{-4}	3.4×10^{-4}	1.7×10^{-7}	5.6	2.6	8.2	4.1×10^{-3}	859	0.34

N/A = not available

a. A collective dose to the 50-mile (80-kilometer) population for atmospheric releases and to the downstream users of the Savannah River for aqueous releases.
b. Arnett and Mamatey (1998) for 1997 data for MEI and population. Worker dose is based on 1997 data (WSRC 1998).
c. Maximum-impacts alternative.
d. DOE (1996); HEU = highly enriched uranium.
e. Less than minimum reportable levels.
f. DOE (1999b).
g. DOE (1994).
h. DOE (1999c).
i. DOE (1998).
j. Derived from impacts from conventional processing of Group A fuel.
k. DOE (1999d).
NRC (1996).

Table 5-5 lists cumulative volumes of high-level, low-level, transuranic, and hazardous and mixed wastes that SRS would generate. The table includes data from the SRS 30-year expected waste forecast (WSRC 1994). The 30-year expected waste forecast is based on operations, environmental restoration, and decontamination and decommissioning waste forecasts from existing generators and the following assumptions: secondary waste from the Defense Waste Processing Facility, In-Tank Precipitation, and Extended Sludge Processing operations are addressed in the DWPF EIS; high-level waste volumes are based on the selected option for the F-Canyon Plutonium Solutions EIS; some investigation-derived wastes are handled as hazardous waste per Resource Conservation and Recovery Act (RCRA) regulations; purge water from well samplings is handled as hazardous waste; and the continued receipt of small amounts of low-level waste from other DOE facilities and nuclear naval operations. The estimated quantity of radioactive/hazardous waste from operations in this forecast during the next 30 years would be 142,666 cubic meters. In addition, radioactive/hazardous waste associated with environmental restoration and decontamination and decommissioning activities would have a 30-year expected forecast of 67,808 cubic meters (Halverson 1999). Waste generated from the conventional processing option would add a total of 154,970 cubic meters. During this same time period, other reasonably foreseeable activities that were not included in the 30-year forecast would add an additional 192,915 cubic. The major contributor to the other waste volumes would be from weapons components from various DOE sites that could be processed in SRS canyons. Therefore, the potential cumulative amount of waste generated from SRS activities during the period of interest would be 558,359 cubic meters. It is important to note that the quantities of waste generated are not equivalent to the amounts that will require disposal. As discussed in Section 4.1.1.4 for example, high-level waste is evaporated and concentrated to a smaller volume for final disposal. Combustible low-level waste is volume reduced on site in the Consolidated Incineration Facility.

The Three Rivers Solid Waste Authority Regional Waste Management Center at the Savannah River Site accepts non-hazardous and non-radioactive solid wastes from SRS and eight surrounding South Carolina counties. This municipal solid waste landfill provides state of the art Subtitle D (non-hazardous) facilities for landfilling solid wastes while reducing the environmental consequences associated with construction and operation of multiple county-level facilities (DOE 1995b). It was designed to accommodate combined SRS and county solid waste disposal needs for at least 20 years, with a projected maximum operational life of 45 to 60 years (DOE 1995b). The landfill is designed to handle an average of 1,000 tons per day and a maximum of 2,000 tons per day of municipal solid wastes. The SRS and eight cooperating counties had a combined generation rate of 900 tons per day in 1995. The Three Rivers Solid Waste Authority Regional Waste Management Center opened in mid-1998.

The SNF management activities and other planned SRS activities would not generate larger volumes of radioactive, hazardous, or solid wastes beyond current and projected capacities of SRS waste storage and/or management facilities.

5.5 Utilities and Energy

Table 5-6 lists the cumulative consumption of electricity from activities at SRS. The values are based on annual consumption estimates. Among the SNF management technologies, Conventional Processing would place the largest annual demand on electricity and water re sources. The SNF management values are based on the maximum impact analysis (Section 4.1.1.5).

The overall SRS activities occurring concurrently with SNF management activities would not place an unreasonable demand on electricity resources.

Table 5-5. Estimated cumulative waste generation from SRS concurrent activities (cubic meters).[a,b,c]

Waste Type	SNF Management[a]	SRS Operations[b,c]	ER/D&D[b,c,d]	Other Waste Volume[e]	Total
High-level	11,000	14,129	0	69,552	94,681
Low-level	140,000	118,669	61,630	110,102	430,401
Hazardous/mixed	270	3,856	6,178	4,441	14,745
Transuranic	3,700	6,012	0	8,820	18,532
Total	154,970	142,666	67,808	192,915	558,359

L4-17 TC

a. Maximum-impact alternative.
b. Halverson (1999).
c. Based on a total 30-year expected waste generation forecast, which includes previously generated waste.
d. ER/D&D = environmental restoration/decontamination & decommissioning.
e. Life-cycle waste associated with reasonably foreseeable future activities such as TEF, plutonium residues, surplus plutonium disposition, highly-enriched uranium, commercial light water reactor waste, sodium-bonded spent nuclear fuel, and weapons components that could be processed in SRS canyons. Impacts for the last group is based on conventional processing impacts for SNF Fuel Group A.

TC EC

Table 5-6. Estimated average annual cumulative utility consumption.

Activity	Electricity (megawatt-hours)	Water usage (liters)
SRS baseline[a]	4.11×10^{5}	1.70×10^{10}
SNF management[b]	1.58×10^{4}	2.11×10^{8}
Other SRS foreseeable activities	1.51×10^{5}	6.73×10^{8}
Total	5.77×10^{5}	1.79×10^{10}

TC

a. Halverson (1999) for electricity usage and Arnett and Mamatey (1996) for water usage.
b. Based on the maximum impact alternative.
c. Includes utility consumption associated with reasonable foreseeable future actions such as tritium extraction, facility, plutonium residues, surplus plutonium disposition, highly-enriched uranium, sodium-bonded spent nuclear fuel, and weapons components that could be processes at SRS canyons. Impacts for last group are based on conventional processing impacts of spent nuclear fuel "Group A." See EISs referenced at end of chapter. Sodium-bonded spent nuclear fuel electricity usage based on "Group A" conventional processing; water usage from EIS.

DOE has also evaluated the SRS water needs during the SNF management activities period. At present, the SRS rate of groundwater with-drawl is estimated to be up to 17 billion liters annually. The estimated amount of groundwater needed for SNF management activities from 1998 to 2035 is 211 million liters per year, depending on the management option chosen. Operation of other foreseeable activities would require approximately 673 million liters of groundwater per year. Thus, sitewide groundwater withdrawals would increase minimally over the projected SNF management period.

TC

Surface water usage during the SNF management period is not projected to approach capacity levels.

TC

5.6 Socioeconomic Impacts

Cumulative regional economic and population changes from construction and operation of the Transfer and Storage Facility or the Transfer, Storage and Treatment Facility consider the impacts of other coincident economic development projects such as DOE's Accelerator for the Pro-

duction of Tritium, Bridgestone-Firestone, and Hankook Synthetics.

Bridgestone-Firestone is building a $435 million tire manufacturing plant in Aiken County that will employ 800 workers. The Bridgestone-Firestone project is expected to complete construction and be in operation by the year 2000. Thus, this project should not impact the construction workforce for the Transfer and Storage Facility or Transfer, Storage and Treatment Facility which are not scheduled to be constructed until after the year 2000. Competition for construction workers should not overlap. TC

Construction of the Transfer and Storage Facility or the transfer and storage phase of the Transfer, Storage and Treatment Facility would begin sometime after the year 2000, employ 500 workers (375 construction and 125 professional), and require 2 years to complete. The treatment phase would begin construction at the completion of the transfer and storage phases and also could employ as many as 500 workers and take as long as 2 years to complete. No additional workers would be required during operations since existing SRS employees would assume those positions. TC

There would be no significant cumulative socioeconomic impacts from construction or operation of the Transfer and Storage Facility or the Transfer, Storage and Treatment Facility.

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CHAPTER 5. CUMULATIVE IMPACTS 1

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Table 5-4. Estimated average annual cumulative radiological doses and resulting health effects to offsite population and facility workers.

	Maximally exposed individual				Offsite population[a]				Workers	
Activity	Dose from airborne releases (rem)	Dose from liquid releases (rem)	Total Dose (rem)	Probability of fatal cancer risk	Collective dose from airborne releases (person-rem)	Collective dose from liquid releases (person-rem)	Total collective dose (person-rem)	Excess latent cancer fatalities	Collective dose	Excess latent cancer fatalities
SRS Baseline[b]	5.0×10^{-5}	1.3×10^{-4}	1.8×10^{-4}	9.5×10^{-8}	2.2	2.4	4.6	2.3×10^{-3}	160	0.066
Management of Spent Nuclear Fuel[c]	1.5×10^{-5}	5.7×10^{-5}	7.2×10^{-5}	3.6×10^{-8}	0.56	0.19	0.75	3.8×10^{-4}	55	0.022
Surplus HEU Disposition[d]	2.5×10^{-6}	(e)	2.5×10^{-6}	1.3×10^{-8}	0.16	(e)	0.16	8.0×10^{-5}	11	4.4×10^{-3}
Tritium Extraction Facility[f]	2.0×10^{-5}	(e)	2.0×10^{-5}	1.0×10^{-8}	0.77	(e)	0.77	3.9×10^{-4}	4	1.6×10^{-3}
Defense Waste Processing Facility[g]	1.0×10^{-6}	(e)	1.0×10^{-6}	5.0×10^{-10}	0.071	(e)	0.071	3.6×10^{-5}	120	0.048
Surplus Plutonium Disposition[h]	4.0×10^{-6}	(e)	4.0×10^{-6}	2.0×10^{-9}	1.6	(e)	1.6	8.0×10^{-4}	541	0.22
Management Plutonium Residues/ Scrub Alloy[i]	2.4×10^{-7}	(e)	2.4×10^{-7}	1.2×10^{-10}	0.026	(e)	0.026	1.3×10^{-5}	25	0.01
DOE complex miscellaneous components[j]	4.4×10^{-6}	4.2×10^{-8}	4.4×10^{-6}	2.2×10^{-9}	7.0×10^{-3}	2.4×10^{-4}	7.2×10^{-3}	3.6×10^{-6}	2	0.001
Sodium-Bonded Spent Nuclear Fuel[k]	3.9×10^{-7}	1.2×10^{-7}	5.1×10^{-7}	2.6×10^{-10}	1.9×10^{-2}	6.8×10^{-4}	2×10^{-2}	9.8×10^{-6}	38	0.015
Plant Vogtle[l]	5.4×10^{-7}	5.4×10^{-5}	5.5×10^{-5}	2.7×10^{-8}	0.042	2.5×10^{-3}	0.045	2.2×10^{-5}	NA	NA
Total	9.8×10^{-5}	2.4×10^{-4}	3.4×10^{-4}	1.7×10^{-7}	5.4	2.6	8.1	4.0×10^{-3}	1,030	0.41

N/A = not available

a. A collective dose to the 50-mile (80-kilometer) population for atmospheric releases and to the downstream users of the Savannah River for aqueous releases.
b. Arnett and Mamatey (1998) for 1997 data for MEI and population. Worker dose is based on 1997 data (WSRC 1998).
c. Maximum-impacts alternative.
d. DOE (1996a); HEU = highly enriched uranium.
e. Less than minimum reportable levels.
f. DOE (1998b, 1999b).
g. DOE (1994).
h. DOE (1998c).
i. DOE (1998a).
j. Derived from impacts from conventional processing of Group A fuel.
k. DOE (1999).
l. NRC (1996).

CHAPTER 6. RESOURCE COMMITMENTS

6.1 Introduction

Chapter 6.0 describes the unavoidable adverse impacts, short-term uses of environmental resources versus long-term productivity, and irreversible or irretrievable commitments of resources associated with safely managing spent nuclear fuel (SNF) at the Savannah River Site (SRS) for the period 1998 to 2035. This chapter also includes discussions about U.S. Department of Energy (DOE) waste minimization, pollution prevention, and energy conservation programs as they would relate to implementation of the proposed action.

6.2 Unavoidable Adverse Impacts

Implementing any of the alternatives considered in this environmental impact statement (EIS) for the management of SNF at SRS would result in minimal unavoidable adverse impacts to the human environment. Construction and operation of a Transfer and Storage Facility to implement the New Packaging Technology or the construction and operation of a Transfer, Storage, and Treatment Facility to implement the New Processing Technology would result in negligible adverse impacts to geologic resources, groundwater, traffic, and cultural resources as described in Chapter 4. All construction activities would occur within the boundary of a reactor or a chemical separations area in an already-developed industrial complex and would require approximately 15 acres.

Potential adverse impacts from construction could occur to surface water resources. However, as part of the required sediment and erosion control plan, storm water management and sediment control measures would minimize runoff from the construction site and potential discharges of silts, solids, and other contaminants to surface-water streams. There would be minimal adverse impacts to air resources from construction activities. Concentrations of pollutants emitted during construction activities would be at least an order of magnitude less than the South Carolina ambient air quality standards concentrations. Likewise, there would be minimal adverse impacts to the ecological resources of the area, primarily due to construction-related noises. Although noise levels would be relatively low outside the immediate area of construction, the combination of construction noise and human activity probably would displace small numbers of animals. These adverse impacts would be small, temporary (24 months or less), and localized. Construction would not disturb any threatened or endangered species, would not degrade any critical or sensitive habitat, and would not affect any jurisdictional wetlands.

Renovating an existing facility for the Transfer, Storage, and Treatment Facility could result in additional low-level waste generation, which could be considered a potential adverse impact. Renovation would require decontamination and removal of components and systems and subsequent construction inside a building, such a reactor building. Adverse impacts would include the generation of approximately 480 m^3 of low-level radioactive waste. This waste volume would have minimal impact on the Site's overall waste management capacity. Eventual decontamination and decommissioning (D&D) of any facility (either new and dedicated to SNF management or renovated to accommodate SNF management) used for the management of SNF would result in the generation of radioactive waste. Impacts of these D&D activities would be evaluated in subsequent National Environmental Policy Act (NEPA) actions.

Unavoidable construction worker radiation exposures would result from renovating an existing reactor facility to become the Transfer, Storage, and Treatment Facility. These occupational exposures (32 person-rem in a population of 54 construction workers) would be well below regulatory limits.

6.3 Relationship between Local Short-Term Uses of the Environment and the Maintenance and Enhancement of Long-Term Productivity

The proposed locations for any new facility are all within developed industrial landscapes. Each of the proposed sites would encompass approximately 15 acres. The existing infrastructure (roads; power-, steam-, and waterlines; wastewater treatment facilities, etc.) within each of the areas is sufficient to support the proposed facilities.

Regardless of location, after the operational life of the project, DOE could decontaminate and decommission (D&D) the facility in accordance with applicable regulatory requirements and restore the area to a brown-field site that would be available for other industrial use. Appropriate NEPA reviews would be conducted prior to the initiation of any D&D action. In all likelihood, none of the sites would be restored to a natural terrestrial habitat.

The project-related uses of environmental resources for the duration of any of the proposed alternatives are characterized below.

- Over the life of the SNF management alternatives, groundwater would be used to meet sanitary and process water needs. After use and treatment, this water would be discharged into surface water streams. Depending on the site chosen and the technology implemented, over the short-term, the resulting increases in pollutant loadings would take advantage of the natural assimilative capacity of the receiving stream(s). However, these incremental pollutant loadings should not adversely affect either short- or long-term productivity of the aquatic ecosystem. These impacts would be assessed during the regulatory permitting process once an alternative has been selected.

- Regardless of location, air emissions associated with implementation of any of the technologies would add small amounts of radiological and nonradiological constituents to the air of the region. During the project's life, these emissions would result in an additional loading and exposure but would not impact SRS compliance with air quality or radiation exposure standards. There would be no significant residual environmental affects to long-term environmental productivity.

- The management and disposal of sanitary solid waste and non-recyclable radiological waste over the project's life would require energy and space at SRS treatment, storage, or disposal facilities (e.g., Three Rivers Sanitary Landfill, E-Area Vaults, Consolidated Incineration Facility). The land required to meet the solid waste needs would require a long-term commitment of terrestrial resources. Upon the facilities' closures, DOE could D&D them and restore them to brown field sites which could be available for future commercial or industrial development.

- Regardless of location, increased employment, expenditures, and tax revenues generated during the implementation of any of the alternatives would directly benefit the local, regional, and state economies over the short-term. Long-term economic productivity could be facilitated by local governments investing project-generated tax revenues into infrastructure and other required services.

6.4 Irreversible and Irretrievable Resource Commitments

Resources that would be irreversibly and irretrievably committed during the implementation of SNF management alternatives include those that cannot be recovered or recycled and those that are consumed or reduced to unrecoverable

forms. The commitment of capital, energy, labor, and material during the implementation of SNF management alternatives would generally be irreversible.

Energy expended would be in the form of fuel for equipment and vehicles, electricity for facility operations, and human labor. Construction would generate nonrecyclable materials such as sanitary solid waste and construction debris. Operation of any proposed facility would generate nonrecyclable waste streams such as radiological and nonradiological solid wastes and some process wastewaters. However, certain materials (e.g., copper, stainless steel) used during construction and operation of the proposed facility could be recycled when the facility was D&Ded. Some construction materials, particularly from existing facilities (e.g., Receiving Basin for Offsite Fuel, L-Reactor Disassembly Area, F- and H-Separation Facilities) would not be salvageable due to radioactive contamination. Table 6-1 lists estimated requirements for concrete and steel for any new facility.

Table 6-2 lists the major materials that would be consumed as a result of process operations, primarily chemicals and other commercial products. Table 2-4 lists the corresponding management technologies that would use the facilities.

The implementation of the SNF management alternatives considered in this EIS, including the No-Action Alternative, would require water, electricity, steam, and diesel fuel. Tables 4.1-15 through 4.1-18 list estimated amounts of these resources that would be consumed during the period of analysis; Section 4.1.1.5 describes the uses. Water would be obtained from onsite groundwater sources and steam from existing onsite sources. Electricity and diesel fuel would be purchased from commercial sources. These commodities are readily available and the amounts required would not have an appreciable impact on available supplies or capacities. From a materials and energy resource commitment perspective, Conventional Processing and the Electrometallurgical Treatment Technology option would recover low enriched uranium, which is useable as commercial reactor fuel. None of the other alternatives would recover this resource.

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6.5 Waste Minimization, Pollution Prevention, and Energy Conservation

6.5.1 WASTE MINIMIZATION AND POLLUTION PREVENTION

DOE has implemented an aggressive waste minimization and pollution prevention program at SRS at the sitewide level and for individual organizations and projects. As a result, significant reductions have been achieved in the amounts of wastes discharged into the environment and sent to landfills, resulting in significant cost savings.

To implement a waste minimization and pollution prevention program at the SNF management facilities, DOE would characterize waste streams and identify opportunities for reducing or eliminating them. Emphasis would be placed on minimizing the largest waste stream, low-level waste, through source reduction and recycling. Selected waste minimization practices could include:

- Process design changes to eliminate the potential for spills and to minimize contamination areas
- Decontamination of equipment to facilitate reuse
- Recycling metals and other usable materials, especially during the construction phase of the project
- Preventive maintenance to extend process equipment life
- Modular equipment designs to isolate potential failure elements to avoid changing out entire units.

L1-5

Table 6-1. Estimated requirements for concrete and steel for stand-alone facilities.

Facility	Concrete (cubic yards)[a]	Steel (tons)[b]
Transfer and Storage Facility (including dry storage vaults)	11,000	600
Transfer, Storage, and Treatment Facility (construction of new facility)	20,000	1,800

a. To convert cubic yards to cubic meters, multiply by 0.764.
b. To convert tons to metric tons, multiply by 0.907.

Table 6-2. Major chemicals and other materials required for spent nuclear fuel management facilities.

Facility	Major material requirements (operation)
Receiving Basin for Offsite Fuel	Water treatment filters, deionizer resins
L-Reactor Disassembly Basin	Water treatment filters, deionizer resins
F or H Canyon	Nitric acid, gelatin, tributyl phosphate, n-paraffin, depleted uranium
Transfer and Storage Facility	Nuclear poison, helium, neutron absorbers, stainless steel (canisters), water treatment filters and deionizer resins (if receipt basin is used)
Melt and Dilute Treatment Facility	Depleted uranium, neutron poison, helium, stainless steel (canisters), glass formers (glass or ceramic frit, silicon dioxide)
Mechanical Dilution Treatment Facility	Depleted uranium, nuclear poison (e.g., borated steel), helium, stainless steel (canisters)
Vitrification Facility	Depleted uranium, glass or ceramic formers (e.g., silicon oxide), stainless steel (canisters), offgas treatment materials (filters, chemicals)
• Dissolve and Vitrify	• Nitric acid, boric acid
• Glass Material Oxidation and Dissolution System	• Boron oxide, lead dioxide (mostly reused in the process), carbon
• Plasma Arc	• Offgas treatment materials (filters, chemicals)
Electrometallurgical Treatment Facility	Depleted uranium; glass; silicon; lithium fluoride, potassium fluoride, and uranium fluoride electrolytes; aluminosilicate filters; waste separation materials (ion exchange media or chemical reduction/oxide precipitation chemicals)

L1-5

- Use of non-toxic or less toxic materials to prevent pollution and minimize hazardous and mixed waste streams

During construction, DOE would implement actions to control surface water runoff and construction debris and to prevent infiltration of contaminants into groundwater. The construction contractor would be selected, in part, based on prior pollution prevention practices.

6.5.2 ENERGY CONSERVATION

SRS has an active energy conservation and management program. Since the mid-1990s more than 40 onsite administrative buildings

have undergone energy efficiency upgrades. Representative actions include the installation of energy-efficient light fixtures, the use of occupancy sensors in rooms, use of diode light sticks in exit signs, and the installation of insulating blankets around hot water heaters. Regardless of location, the incorporation of these types of energy-efficient technologies into facility de-sign, along with the implementation of process efficiencies and waste minimization concepts, would facilitate energy conservation by any of the SNF management alternatives.

CHAPTER 6. RESOURCE COMMITMENTS 1

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CHAPTER 7. APPLICABLE LAWS, REGULATIONS, AND OTHER REQUIREMENTS

This chapter identifies and summarizes the major laws, regulations, Executive Orders, and Department of Energy (DOE) Orders that could apply to the management of spent nuclear fuel (SNF) at the Savannah River Site (SRS). Permits or licenses could be required under some of these laws and regulations. However, DOE would determine the specific requirements for permits or licenses, which would depend greatly on the chosen alternative, after consultation with the appropriate regulating agencies.

Section 7.1 discusses the major Federal and State of South Carolina statutes and regulations that impose environmental protection requirements on DOE and which require DOE to obtain a permit prior to construction and operation of spent nuclear fuel facilities. Each of the applicable regulations establishes how potential releases of pollutants and radioactive materials are to be controlled or monitored and include requirements for the issuance of permits for new operations or new emission sources. In addition to environmental permit requirements, the statutes may require consultations with various authorities to determine if an action (such as construction and operation of a facility) requires a permit or the implementation of protective or mitigative measures. Sections 7.1.1 and 7.1.2 discuss the environmental permitting process and lists the environmental permits and consultations (see Table 7-1) applicable to construction and operation of the spent nuclear fuel facilities.

Sections 7.2 and 7.3 address the major Federal regulations and Executive Orders, respectively, which address issues such as protection of public health and the environment, worker safety, and emergency planning. The Executive Orders clarify issues of national policy and set guidelines under which Federal agencies must act.

DOE implements its responsibilities for protection of public health, safety, and the environment through a series of Departmental Orders (see Section 7.4) that are mandatory for operating contractors of DOE-owned facilities.

7.1 Statutes and Regulations Requiring Permits or Consultations

Environmental regulations require that the owner or operator of a facility obtain permits for the construction and operation of new (water and air) emissions sources, and for new domestic drinking water systems. To obtain these permits, the facility operator must apply to the appropriate government agency for a discharge permit for discharges of wastewater to the waters of the state and submit construction plans and specifications for the new emission sources, including new air sources. The environmental permits contain specific conditions with which the permittee must comply during construction and operation of a new emission source, describe pollution abatement and prevention methods to be utilized for reduction of pollutants, and contain emissions limits for pollutants which will be emitted from the facility. Section 7.1.1 discusses the environmental statutes and regulations under which DOE will be required to obtain permits. Table 7-1 lists the permits.

7.1.1 ENVIRONMENTAL PROTECTION PERMITS

Clean Air Act, as amended, (42 USC 7401 et seq.), (40 CFR Parts 50-99); South Carolina Pollution Control Act [Section 48-1-30 et seq., South Carolina Department of Health and Environmental Control (SCDHEC) Regulation 61-62]

The Clean Air Act, as amended, is intended to "protect and enhance the quality of the Nation's air resources so as to promote the public health and welfare and the productive capacity of its

population." Section 118 of the Clean Air Act, as amended, requires each Federal agency, such as DOE, with jurisdiction over any property or facility that might result in the discharge of air pollutants, to comply with "all Federal, State, interstate, and local requirements" with regard to the control and abatement of air pollution.

The Act requires the U.S. Environmental Protection Agency (EPA) to establish National Ambient Air Quality Standards as necessary to protect public health, with an adequate margin of safety, from any known or anticipated adverse effects of a regulated pollutant (42 USC 7409). The Act also requires the establishment of national standards of performance for new or modified stationary sources of atmospheric pollutants (42 USC 7411) and requires specific emission increases to be evaluated so as to prevent a significant deterioration in air quality (42 USC 7470). Hazardous air pollutants, including radionuclides, are regulated separately (42 USC 7412). Air emissions are regulated by the EPA in 40 CFR Parts 50 through 99. In particular, radionuclide emissions are regulated under the National Emission Standard for Hazardous Air Pollutants Program (NESHAP) (see 40 CFR Part 61).

EPA has overall authority for the Clean Air Act; however, it delegates primary authority to states which have an established air pollution control program approved by EPA. In South Carolina, EPA has retained authority over radionuclide emissions (40 CFR Part 61) and has delegated to SCDHEC the responsibility for the rest of the regulated pollutants under the authority of the South Carolina Pollution Control Act (48-1-10 et. seq.,) and SCDHEC Air Pollution Control Regulations 61-62.

Construction and operation permits or exemptions will be required for new nonradiological air emission sources (diesel generators, concrete batch plants etc.) constructed and operated at any SNF facility. The permits will contain operating conditions and effluent limitations for pollutants emitted from the facilities (see Table 7-1).

DOE is currently determining if a NESHAP permit will be required for radiological emissions from any spent nuclear fuel facilities (stacks, process vents, etc.). As described in 40 CFR Part 61.96, if the effective dose equivalent caused by all emissions from facility operations is projected to be less than 1 percent of the 10 millirem per year NESHAP standard, an application for approval to construct under 40 CFR Part 61.07 is not required to be filed. 40 CFR Part 61.96 also allows DOE to use, with prior EPA approval, methods other than EPA standard methods for estimating the source term for use in calculating the projected dose. DOE is currently investigating methods for estimating the transfer, storage and treatment facility source term in accordance with NESHAP requirements to calculate if the emissions would result in an effective dose equivalent of less than the 0.1 millirem per year level. Based on the results of this calculation, DOE will, prior to the start of construction, request EPA approval of the methodology for calculating the projected dose or complete a NESHAP permit application.

Federal Clean Water Act, as amended (33 USC 1251 et seq.); SC Pollution Control Act (SC Code Section 48-1-10 et seq., 1976) (SCDHEC Regulation 61-9.122 et. seq.)

The Federal Water Pollution Act (commonly known as the Clean Water Act), was enacted to "restore and maintain the chemical, physical and biological integrity of the Nation's water." The Clean Water Act prohibits the "discharge of toxic pollutants in toxic amounts" to navigable waters of the United States (Section 101). Section 313 of the Clean Water Act, as amended, requires all branches of the Federal Government engaged in any activity that might result in a discharge or runoff of pollutants to surface waters to comply with Federal, state, interstate, and local requirements.

In addition to setting water quality standards for the Nation's waterways, the Clean Water Act supplies guidelines and limitations (Sections 301-303) for effluent discharges from

point-source discharges and provides authority (Sections 401-402) for the EPA to implement the National Pollutant Discharge Elimination System (NPDES) permitting program pursuant to 40 CFR Part 122 *et seq.*

EPA has delegated primary enforcement authority for the Clean Water Act and the NPDES Permitting Program to SCDHEC for waters in South Carolina. In 1996, SCDHEC, under the authority of the Pollution Control Act (48-1-10 et seq.) and Regulation 61-9.122, issued NPDES Permit SC0000175, which addresses wastewater discharges to SRS streams and NPDES permit SCG250162 which address general utility water discharges. The permit contains effluent limitations for physical parameters such as flow and temperature and for chemical pollutants with which the permittee/discharge must comply. DOE will apply for a discharge permit for SNF facilities if the process chosen results in discharges to waters of the State (see Table 7-1).

In Section 402(p) of the Clean Water Act EPA established regulations (40 CFR Part 122.26) for issuing permits for stormwater discharges associated with industrial activity. Accordingly, SCDHEC has issued a General Permit for Storm Water Discharges Associated with Industrial Activities (Permit No. SCR000000) authorizing stormwater discharges to the waters of the State of South Carolina in accordance with effluent limitations, monitoring requirements, and conditions as set forth in the permit. This permit requires preparation and submittal of a Pollution Prevention Plan for all new and existing point source discharges associated with industrial activity. Accordingly, DOE-SR has developed a Storm Water Pollution Prevention Plan (SWPPP) for storm water discharges at SRS. The SRS SWPPP would need to be revised to include pollution prevention measures to be implemented for operation of SNF facilities (See Table 7-1) if industrial activities are exposed to stormwater. SCDHEC has issued a General Permit for stormwater discharges from construction activities that are "Associated with Industrial Activity" (Permit No. SCR100000). An approved plan would be needed that includes erosion control and pollution prevention measures to be implemented for construction activities.

Section 404 of the Clean Water Act requires that a 404 Permit be issued for discharge of dredge or fill material into the waters of the United States. The authority to implement these requirements has been given to the U.S. Army Corps of Engineers. Section 401 of the Clean Water Act requires certification that discharges from construction or operation of facilities, including discharges of dredged and fill material into navigable waters will comply with applicable water standards. This certification, which is granted by SCDHEC, is a prerequisite for the 404 permit. DOE does not believe that a 404 permit will be required for construction of the SNF facilities.

Federal Safe Drinking Water Act, as amended [42 USC 300 (F) et seq., 40 CFR Parts 100-149]; South Carolina Safe Drinking Water Act (Title 44-55-10 et seq.), State Primary Drinking Water Regulations, (SCDHEC R.61-58)

The primary objective of the Safe Drinking Water Act (42 USC 300), as amended, is to protect the quality of the public water supplies and all sources of drinking water. The implementing regulations, administered by the EPA unless delegated to the States, establish standards applicable to public water systems. They promulgate maximum contaminant levels (including those for radioactivity), in public water systems, which are defined as water systems that serve at least 15 service connections used by year-round residents or regularly serve at least 25 year-round residents. Safe Drinking Water Act requirements have been promulgated by the EPA in 40 CFR Parts 100 through 149. Other programs established by the Safe Drinking Water Act include the Sole Source Aquifer Program, the Wellhead Protection Program, and the Underground Injection Control Program.

EPA has delegated primary enforcement authority to SCDHEC for public water systems in South Carolina. Under the authority of the South Carolina Safe Drinking Water Act (44-55-10 et

seq.), SCDHEC has established a drinking water regulatory program (R.61-58). For radioactive material, the regulations specify that the average annual concentration of manmade radionuclides in drinking water as delivered to the user by such a system shall not produce a dose equivalent to the total body or an internal organ greater than four millirem per year beta-gamma activity. Construction and operation permits will be required for any major new components associated with the SNF facilities. See Table 7-1.

Resource Conservation and Recovery Act, as amended (Solid Waste Disposal Act) (42 USC 6901 et seq.); South Carolina Hazardous Waste Management Act, Section 44-56-30, South Carolina Hazardous Waste Management Regulations (R.61-79.124 et seq.)

The treatment, storage, or disposal of hazardous and nonhazardous waste is regulated under the Solid Waste Disposal Act as amended by the Resource Conservation and Recovery Act (RCRA) and the Hazardous and Solid Waste Amendments of 1984. Pursuant to Section 3006 of the Act, any state that seeks to administer and enforce a hazardous waste program pursuant to RCRA may apply for Environmental Protection Agency authorization of its program. The EPA regulations implementing RCRA (40 CFR Parts 260 through 280) define hazardous wastes and specify their transportation, handling, treatment, storage, and disposal requirements.

The regulations imposed on a generator or a treatment, storage, or disposal facility vary according to the type and quantity of material or waste generated, treated, stored, or disposed of. The method of treatment, storage, or disposal also affects the extent and complexity of the requirements.

Historically, DOE chemically processed spent nuclear fuel to recover valuable products and fissionable materials, and as such, the spent nuclear fuel was not a solid waste under the Resource Conservation and Recovery Act.

World events have resulted in significant changes in DOE's direction and operations. In particular, in April 1992, DOE announced the phase-out of processing for the recovery of special nuclear materials. With these changes, DOE's focus on most of its spent nuclear fuel has changed from processing and recovery of materials to storage and ultimate disposition. This in turn has created uncertainty regarding the regulatory status of some of DOE's spent nuclear fuel relative to the Resource Conservation and Recovery Act.

DOE has initiated discussion with the Environmental Protection Agency on the potential applicability of the Resource Conservation and Recovery Act to spent nuclear fuel. In 1995, an investigation of the applicability of RCRA regulations to a variety of spent fuels and special nuclear materials that were then stored on the SRS (Huggins 1995) concluded, based largely on process knowledge and engineering judgment, that none of the spent fuel in question contained RCRA listed or characteristic material. The evaluated fuel types and cladding materials included aluminum cladding, uranium metal, thorium dioxide, uranium and thorium dioxide powders and pellets, uranium-plutonium powders and pellets, and beryllium oxide powders and pellets. The specific fuels are not necessarily identical to those evaluated in this EIS; however the calculations were conservative and assumed that similar types of fuel or cladding would generally have the same material specifications regardless of where the fuel was fabricated (Huggins 1995). Uranium silicide fuels were not considered in the 1995 evaluation but, based on the general chemical composition (Knight 1993) of uranium silicide fuel and the method used for Toxicity Characteristic Leaching Procedure (TCLP) calculations (Huggins 1995), it does not appear that uranium silicide fuels would qualify as a RCRA hazardous waste.

The Federal Facility Compliance Act (FFCA) (42 USC 6921 (et. seq.)

The FFCA was enacted on October 6, 1992, amended the Resource Conservation Recovery Act. The FFCA waived sovereign immunity for fines and penalties for violations at Federal facilities associated with the management of mixed waste. However, a provision postpones fines and penalties after 3 years for mixed waste storage prohibition violations at DOE sites and requires DOE to prepare plans for developing the required treatment capacity for mixed waste stored or generated at each facility. Each plan must be approved by the host State or the EPA, after consultation with other affected States, and a consent order must be issued by the regulator requiring compliance with the plan. The Federal Facility Compliance Act further provides that DOE will not be subject to fines and penalties for land disposal restriction storage prohibition violations for mixed waste as long as it is in compliance with such an approved plan and consent order and meets all other applicable regulations. This would apply to mixed waste generated as a result of operation of SNF management facilities which are subject to requirements of the Resource Conservation and Recovery Act. On September 20, 1995, the SCDHEC approved, with modification, the Site Treatment Plan for SRS. SCDHEC issued a consent order, signed by DOE, requiring compliance with the plan on September 29, 1995. DOE would be required to notify SCDHEC of new mixed waste streams generated as result of SNF management operations.

Federal Aviation Act of 1958 (49 USC 1504) Federal Aviation Administration Regulations (14 CFR Part 77)

The Federal Aviation Administration requires that a permit be issued for any structure greater than 200 feet in height which would affect navigable airspace (see Table 7-1). A permit would be required for structures at the SNF site greater than 200 feet in height.

7.1.2 PROTECTION OF BIOLOGICAL, HISTORIC, AND ARCHAEOLOGICAL RESOURCES

The following statutes pertain to protection of endangered and threatened animal and plants.

Endangered Species Act, as amended (16 USC 1531 et seq.)

The Endangered Species Act, as amended, is intended to prevent the further decline of endangered and threatened species and to restore these species and their habitats. The Act is jointly administered by the United States Departments of Commerce and Interior. Section 7 of the Act requires consultation with the Fish and Wildlife Service (Interior) and the National Marine Fisheries Service (Commerce) to determine if endangered and threatened species or their critical habitats are in the vicinity of the proposed action. DOE will comply with the Section 7 Process.

All sites considered for construction of new SNF management facilities are within fenced, disturbed industrial areas. The potential for conditions suitable to support threatened or endangered species does not exist.

Migratory Bird Treaty Act, as amended (16 USC 703 et seq.)

The Migratory Bird Treaty Act, as amended, is intended to protect birds that have common migration patterns between the United States and Canada, Mexico, Japan, and Russia. It regulates the harvest of migratory birds by specifying things such as the mode of harvest, hunting seasons, and bag limits. The Act stipulates that it is unlawful at any time, by any means, or in any manner to "kill...any migratory bird." DOE would be required to consult with the Fish and Wildlife Service regarding impacts to migratory birds and to evaluate ways to avoid or minimize these effects in accordance with the Fish and Wildlife Service Mitigation Policy during construction and operation of SNF management facilities.

Bald and Golden Eagle Protection Act, as amended (16 USC 668-668d)

The Bald and Golden Eagle Protection Act makes it unlawful to take, pursue, molest, or disturb bald and golden eagles, their nests, or their eggs anywhere in the United States (Sections 668, 668c). A permit must be obtained from the U.S. Department of the Interior to relocate a nest that interferes with resource development or recovery operations. All sites considered for the SNF management facilities are within fenced industrial areas without habitat suitable for nesting eagles.

National Historic Preservation Act, as amended (16 USC 470 et seq.)

The National Historic Preservation Act, as amended, provides that sites with significant national historic value be placed on the *National Register of Historic Places*. No permits or certifications are required under the Act. However, if a particular Federal activity could impact an historic property resource, consultation with the Advisory Council on Historic Preservation will usually generate a Memorandum of Agreement, including stipulations that must be followed to minimize adverse impacts. Coordination with the South Carolina State Historic Preservation Officer (SC SHPO) ensures the proper identification of potentially significant sites and the implementation of appropriate mitigative actions. All sites considered for SNF management facilities are within previously disturbed industrial sites.

Archaeological Resource Protection Act, as amended (16 USC 470 et seq.)

This Act requires a permit for any excavation or removal of archaeological resources from public or Native American lands. Excavations must be undertaken for the purpose of furthering archaeological knowledge in the public interest, and resources removed are to remain the property of the United States. Consent must be obtained from the Indian Tribe owning lands on which a resource is located before a permit is issued, and the permit must contain terms or conditions requested by the Tribe.

Native American Grave Protection and Repatriation Act of 1990 (25 USC 3001)

This law directs the Secretary of Interior to assume responsibilities for repatriation of Federal archaeological collections and collections held by museums receiving Federal funding that are culturally affiliated with Native American Tribes. Major actions to be taken under this law include (1) establishing a review committee with monitoring and policy-making responsibilities, (2) developing regulations for repatriation, including procedures for identifying lineal descent or cultural affiliation needed for claims, (3) overseeing museum programs designed to meet the inventory requirements and deadlines of this law, and (4) developing procedures to handle unexpected discoveries of graves or grave goods during activities on Federal or tribal land.

American Indian Religious Freedom Act of 1978 (42 USC l996)

This Act reaffirms Native American religious freedom under the First Amendment, and sets U.S. policy to protect and preserve the inherent and constitutional right of Native Americans to believe, express, and exercise their traditional religions. The Act requires that Federal actions avoid interfering with access to sacred locations and traditional resources that are integral to the practice of religion.

In conjunction with 1991 studies related to the New Production Reactor, DOE solicited the concerns of Native Americans about 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 general concerns about SRS and the Central Savannah River Area, but did not identify specific sites as possessing religious significance. The Yuchi Tribal Organization and the National Council of Muskogee Creek are interested in plant species traditionally used in tribal ceremonies, such as redroot, button snakeroot, and American ginseng (DOE 1991). Redroot and

button snakeroot are known to occur on the SRS (Batson, Angerman, and Jones 1985).

In addition, the Savannah River Archaeological Research Program (SRARP) conducted an archeological survey of the preferred APT site in March 1997. The archeological review included potential sites associated with Native American activities or habitat. The resulting SRARP report stated that no archaeological sites present on the preferred site were eligible for nomination to the National Registry of Historical Places and further indicated that SRARP would request from the SC SHPO a determination of no effect from the construction of APT at the preferred site.

7.2 Statutes and Regulations Related to Emergency Planning, Worker Safety, and Protection of Public Health and the Environment

7.2.1 ENVIRONMENTAL PROTECTION

National Environmental Policy Act (NEPA) of 1969, as amended (42 USC 4321 et seq.)

NEPA establishes a national policy promoting awareness of the environmental consequences of human activity on the environment and consideration of environmental impacts during the planning and decisionmaking stages of a project. This Act requires Federal agencies to prepare a detailed statement on the environmental effects of proposed major Federal actions that might significantly affect the quality of the human environment.

This EIS has been prepared in response to NEPA requirements and policies, and in accordance with Council on Environmental Quality (40 CFR Parts 1500 through 1508) and DOE (10 CFR Part 1021) regulations for implementing the procedural provisions of NEPA. It discusses reasonable alternatives and their potential environmental consequences.

Pollution Prevention Act of 1990 (42 USC 13101 et seq.)

The Pollution Prevention Act of 1990 establishes a national policy for waste management and pollution control that focuses first on source reduction, followed sequentially by environmentally safe recycling, treatment, and disposal. Disposal or releases to the environment should occur only as a last resort. In response, DOE has committed to participation in the Superfund Amendments and Reauthorization Act Section 313, U.S. EPA 33/50 Pollution Prevention Program. The goal for facilities already involved in Section 313 compliance is to achieve by 1997 a 33-percent reduction in the release of 17 priority chemicals from a 1993 baseline. On August 3, 1993, President Clinton issued Executive Order 12856, expanding the 33/50 program such that DOE had to reduce its total releases of all toxic chemicals by 50 percent by December 31, 1999. In addition, DOE is requiring each of its sites to establish site-specific goals to reduce the generation of all waste types. EC

Comprehensive Guideline for Procurement of Products Containing Recovered Materials (40 CFR Part 247)

This regulation is issued under the authority of Section 6002 of the Resource Conservation and Recovery Act and Executive Order 12783, which set forth requirements for Federal agencies to procure products containing recovered materials for use in their operations using guidelines established by the EPA. The purpose of these regulations is to promote recycling by using government purchasing to expand markets for recovered materials. RCRA Section 6002 requires that any purchasing agency, when using appropriated funds to procure an item, shall purchase it with the highest percentage of recovered materials practicable. The procurement of materials to be utilized in the construction and operation of SNF management facilities should be conducted in accordance with these regulations.

Toxic Substances Control Act, as amended (USC 2601 et seq.) (40 CFR Part 700 et seq.)

The Toxic Substances Control Act regulates the manufacture, use, treatment, storage, and disposal of certain toxic substances not regulated by the Resource Conservation and Recovery Act or other statutes, particularly polychlorinated biphenyls (40 CFR Part 761), chlorofluorocarbons (40 CFR Part 762), and asbestos (40 CFR Part 763). It is expected that the use of these materials at APT would be limited, or not occur; however, programs and procedures would need to be implemented to address appropriate management and disposal of waste generated as a result of their use.

7.2.2 EMERGENCY PLANNING AND RESPONSE

This section discusses the regulations which address protection of public health, worker safety, and require the establishment of emergency plans and the coordination with local and Federal agencies related to facility operations. DOE Orders generally set forth the programs and procedures required to implement the requirements of these regulations. See Section 7.4.

Atomic Energy Act of 1954, as amended (42 USC 2011 et seq.)

The Atomic Energy Act of 1954 authorizes DOE to establish standards to protect health or minimize dangers to life or property with respect to activities under its jurisdiction. Through a series of Orders, DOE has established an extensive system of standards and requirements to ensure the safe operation of its facilities.

Atomic Energy Act of 1954, as amended (42 USC 2011 et seq.) Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release (10 CFR Part 30.72 Schedule C)

This list is the basis for both the public and private sector to determine if the radiological materials they deal with must have an emergency response plan for unscheduled releases. It is one of the threshold criteria documents for DOE Emergency Preparedness Hazards Assessments required by DOE Order 151.1, "Comprehensive Emergency Management System." An emergency response plan addressing SNF operations would need to promulgated in accordance with this regulation.

Reorganization Plan No. 3 of 1978, Public Health and Welfare (42 USC 5121 et seq.), Emergency Management and Assistance (44 CFR Part 1-399)

These regulations generally include the policies, procedures and set forth the responsibilities of the Federal Emergency Management Agency, the Nuclear Regulatory Commission, and the Department of Energy for implementing a Federal Emergency Preparedness Program including radiological planning and preparedness. An emergency response plan, including radiological planning and preparedness for SNF management operations, would need to prepared and implemented, in accordance with this regulation.

Emergency Planning and Community Right-to-Know Act of 1986 (42 USC 11001 et seq.) (also known as "SARA Title III")

The Emergency Planning and Community Right-to-Know Act of 1986 requires emergency planning and notice to communities and government agencies of the presence and release of specific chemicals. EPA implements this Act under regulations found at 40 CFR Parts 355, 370, and 372. Under Subtitle A of this Act, Federal facilities provide various information (such as inventories of specific chemicals used or stored and releases that occur from these facilities) to the State Emergency Response Commission and the Local Emergency Planning Committee to ensure that emergency plans are sufficient to respond to unplanned releases of hazardous substances. Implementation of the provisions of this Act began voluntarily in 1987, and inventory and annual emissions reporting began in 1988. In addition, DOE requires compliance with Title III as a matter of Departmental policy. The requirements for this Act were promulgated by EPA in 40 CFR Parts 350 through 372. The SRS submits hazardous chemical inventory re-

ports to the SCDHEC. The chemical inventory could change depending on the alternative(s) DOE implemented; however, subsequent reports would reflect any change to the inventory.

Transportation of Hazardous Materials (49 USC 5101 et seq.); Hazardous Materials Tables & Communications, Emergency Response Information Requirements (49 CFR Part 172)

The regulatory requirements for marking, labeling, placarding, and documenting hazardous materials shipments are defined in this regulation. It also specifies the requirements for providing hazardous material information and training. Materials shipped to and from SNF management facilities would be required to comply with these regulations.

Comprehensive Environmental Response, Compensation, and Liability Act of 1980, as amended (42 USC 9601 et seq.) National Oil and Hazardous Substance Contingency Plan (40 CFR Part 300 et seq.)

More popularly known as "Superfund," the Act and implementing regulations provide the needed general authority for Federal and state governments to respond directly to hazardous substances incidents. The regulations require reporting of spills, including radioactive, to the National Response Center. SNF management operations would be required to comply with these regulations in the event of spills of hazardous materials at SNF facilities. DOE Orders generally set forth the programs for development of internal procedures for implementing the regulations.

Occupational Safety and Health Act of 1970, as amended (29 USC 651 et seq.); Occupational Safety and Health Administration Emergency Response, Hazardous Waste Operations and Worker Right to Know (29 CFR Part 1910 et seq.)

The Occupational Safety and Health Act (29 USC 651) establishes standards to enhance safe and healthful working conditions in places of employment throughout the United States. The Act is administered and enforced by the Occupational Safety and Health Administration, a U.S. Department of Labor agency. While the Occupational Safety and Health Administration and EPA both have a mandate to reduce exposures to toxic substances, the Occupational Safety and Health Administration's jurisdiction is limited to safety and health conditions that exist in the workplace environment. In general, under the Act, it is the duty of each employer to furnish all employees a place of employment free of recognized hazards likely to cause death or serious physical harm. Employees have a duty to comply with the occupational safety and health standards and all rules, regulations, and orders issued under the Act. The Occupational Safety and Health Administration regulations (29 CFR) establish specific standards telling employers what must be done to achieve a safe and healthful working environment. This regulation sets down the Occupational Safety and Health Administration requirements for employee safety in a variety of working environments. It addresses employee emergency and fire prevention plans (Section 1910.38), hazardous waste operations and emergency response (Section 1910.120), and hazards communication (Section 1910.1200) that enables employees to be aware of the dangers they face from hazardous materials at their workplace. DOE places emphasis on compliance with these regulations at its facilities and prescribes through DOE Orders the Occupational Safety and Health Act standards that contractors shall meet, as applicable to their work at Government-owned, contractor-operated facilities. DOE keeps and makes available the various records of minor illnesses, injuries, and work-related deaths required by Occupational Safety and Health Administration regulations.

Noise Control Act of 1972, as amended (42 USC 4901 et seq.)

Section 4 of the Noise Control Act of 1972, as amended, directs all Federal agencies to carry out "to the fullest extent within their authority" programs within their jurisdictions in a manner that furthers a national policy of promoting an envi-

ronment free from noise that jeopardizes health and welfare.

7.3 Executive Orders

The following executive orders would be in effect for the construction and operation of the APT. DOE Orders generally set forth the programs and procedures required to implement the requirements of the orders.

Executive Order 11514 (Protection and Enhancement of Environmental Quality)

Executive Order 11514 requires Federal agencies to monitor and control their activities continually to protect and enhance the quality of the environment and to develop procedures to ensure the fullest practicable provision of timely public information and understanding of Federal plans and programs with environmental impact to obtain the views of interested parties.

Executive Order 11988 (Floodplain Management)

Executive Order 11988 requires Federal agencies to establish procedures to ensure that the potential effects of flood hazards and floodplain management are considered for any action undertaken in a floodplain and that floodplain impacts be avoided to the extent practicable.

Executive Order 11990 (Protection of Wetlands)

Executive Order 11990 requires Government agencies to avoid any short- and long-term adverse impacts on wetlands wherever there is a practicable alternative.

Executive Order 12856 (Right-to-Know Laws and Pollution Prevention Requirements)

Executive Order 12856 requires all Federal agencies to reduce the toxic chemicals entering any waste stream. This order also requires Federal agencies to report toxic chemicals entering waste streams; improve emergency planning, response, and accident notification; and encourage clean technologies and testing of innovative prevention technologies.

Executive Order 12898 (Environmental Justice)

Executive Order 12898 requires Federal agencies to identify and address disproportionately high and adverse human health or environmental effects of its programs, policies, and activities on minority and low-income populations.

Executive Order 12902 (Energy Efficiency and Water Conservation at Federal Facilities)

Executive Order 12902 requires Federal agencies to develop and implement a program for conservation of energy and water resources.

7.4 DOE Regulations and Orders

Through the authority of the Atomic Energy Act, DOE is responsible for establishing a comprehensive health, safety, and environmental program for its facilities. The regulatory mechanisms through which DOE manages its facilities are the promulgation of regulations and the issuance of DOE Orders. Table 7-2 lists the major DOE Orders applicable to the construction and operation of SNF management facilities.

The DOE regulations address such areas as energy conservation, administrative requirements and procedures, nuclear safety, and classified information. For the purposes of this EIS, relevant regulations include 10 CFR Part 820, *Procedural Rules for DOE Nuclear Facilities*; 10 CFR Part 830, *Nuclear Safety Management; Contractor and Subcontractor Activities*; 10 CFR Part 835, *Occupational Radiation Protection*; 10 CFR Part 1021, *Compliance with NEPA*; and 10 CFR Part 1022, *Compliance with Floodplains/Wetlands Environmental Review Requirements*. DOE has enacted occupational radiation protection standards to protect DOE and its contractor employees. These standards are set forth in 10 CFR Part 835, *Occupational Radiation Protection*; the rules in this part establish radiation protection standards, limits, and

program requirements for protecting individuals from ionizing radiation resulting from the conduct of DOE activities, including those conducted by DOE contractors. The activity may be, but is not limited to, design, construction, or operation of DOE facilities. These regulations would be in effect for the construction and operation of any facilities associated with the production and management of tritium. DOE Orders generally set forth policy and the programs and internal procedures for implementing those policies.

Table 7-2. DOE Orders and Notices relevant to spent nuclear fuel management.

DOE Order	DOE Orders
151.1	Comprehensive Emergency Management System
225.1	Accident Investigations
231.1	Environment, Safety, and Health Reporting
232.1	Occurrence Reporting and Processing of Operations Information
420.1	Facility Safety
425.1	Startup and Restart of Nuclear Facilities
430.1	Life-Cycle Asset Management
440.1	Worker Protection Management for DOE Federal and Contractor Employees
441.1	DOE Radiological Health and Safety Policy
441.2	Extension of DOE 441.1 (9-19-96)
441.3	Extension of DOE 441.1 (9-17-97)
451.1A	National Environmental Policy Act Compliance Program
460.1A	Packaging and Transportation Safety
460.2	Departmental Materials and Packaging Management
470.1	Safeguards and Security Program
471.1	Identification and Protection of Unclassified Controlled Nuclear Information
471.2A	Information Security Program
472.1B	Personnel Security Activities
1270.2B	Safeguards Agreement with the International Atomic Energy Agency
1300.2A	Department of Energy Technical Standards Program
1360.2B	Unclassified Computer Security Program
3790.1B	Federal Employee Occupational Safety and Health Program
4330.4B	Maintenance Management Program
4700.1	Project Management System
5400.1	General Environmental Protection Program
5400.3	Hazardous and Radioactive Mixed Waste Program
5400.5	Radiation Protection of the Public and the Environment
5480.4	Environmental Protection, Safety, and Health Protection Standards
5480.17	Site Safety Representatives
5480.19	Conduct of Operations Requirements for DOE Facilities
5480.20A	Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities
5480.21	Unreviewed Safety Questions
5480.22	Technical Safety Requirements
5480.23	Nuclear Safety Analysis Reports
5480.25	Safety of Accelerator Facilities
5480.27	Equipment Qualification for Reactor and Nonreactor Nuclear Facilities
5484.1	Environmental Protection, Safety, and Health Protection Information Reporting Requirements
5630.12A	Safeguards and Security Inspection and Evaluation Program
5632.1C	Protection and Control of Safeguards and Security Interests
5633.3B	Control and Accountability of Nuclear Materials
5660.1B	Management of Nuclear Materials
5700.6C	Quality Assurance
5820.2A	Radioactive Waste Management
6430.1A	General Design Criteria

Table 7-2. (Continued)

	DOE Standards
1020-94	Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities
1021-94	Natural Phenomena Hazards Performance Categorization Criteria for Structure, Systems, and Components
1024-92	Guidelines for Use of Probabilistic Seismic Hazard Curves at Department of Energy Sites
1027-92	Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23 Nuclear Safety Analysis Reports
3009-94	Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports
3011-94	DOE Standard Guidance for Preparation of DOE 5480.22 and DOE 5480.23 Implementation Plans

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DOE (U.S. Department of Energy), 1991, American Indian Religious Freedom Act Compliance at the Savannah River Site, Savannah River Operations Office, Aiken, South Carolina.

Huggins, L. B., 1995, *RCRA Characterization of Nuclear Materials in Inventory (NMII) Within the Excess Facilities and Reactor Fuel Storage Program Division*, Westinghouse Savannah River Company, Aiken, South Carolina.

Knight, R. W., 1993, *Observations in the Manufacture of Aluminum-Based Research Reactor Fuel Elements*, Martin Marietta, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

CHAPTER 7. APPLICABLE LAWS, REGULATIONS, AND OTHER REQUIREMENTS ……1

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Table 7-1. Environmental permits and consultations required by law.

Activity/Topic	Law	Requirements	Agency
Site Preparation	Federal Clean Water Act (Section 404)	Stormwater Pollution Prevention Plan for Industrial Activity	SCDHEC[a]
Wastewater Discharges	Federal Clean Water Act S.C. Pollution Control Act	Stormwater Pollution Prevention/Erosion Control Plan for construction activity	SCDHEC WSRC/EPD[b]
		NPDES Permit(s) for Process Wastewater Discharges	SCDHEC
		Process Wastewater Treatment Systems Construction and Operation Permits (if applicable)	SCDHEC
		Sanitary Waste Water Pumping Station Tie-in Construction Permit; Permit to Operate	SCDHEC WSRC/EPD
Air	Clean Air Act - NESHAP	Rad Emissions - Permit to construct new emission source (if needed)	EPA[c]
		Air Construction and Operation permits - as required (e.g., Fire Water Pumps; Diesel Generators)	SCDHEC
		General source - Stacks, Vents, Concrete batch plant	SCDHEC
		Air Permit - Prevention of Significant Deterioration (PSD)	SCDHEC
Domestic Water	Safe Drinking Water Act	Construction and operation permits for line to domestic water system	WSRC/EPD SCDHEC
Endangered Species	Endangered Species Act	Consultation	U.S. Fish and Wildlife Service; National Marine Fisheries Service
Migratory Birds	Migratory Bird Treaty Act	Consultation	U.S. Fish and Wildlife Service

a. South Carolina Department of Health and Environmental Control.
b. WSRC/EPD Westinghouse Savannah River Company Environmental Protection Department.
c. Environmental Protection Agency.

DOE/EIS-0279 Figures

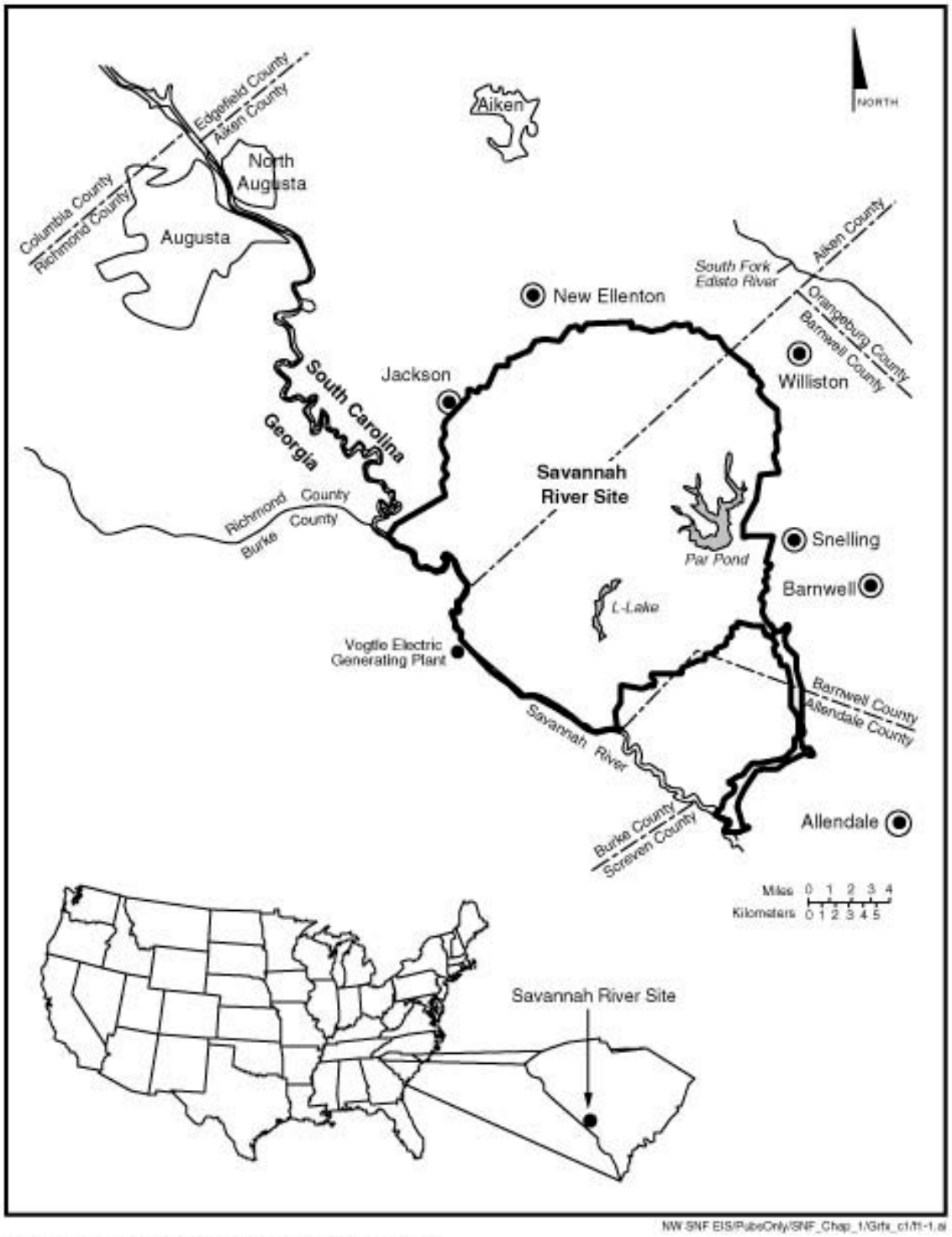

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Figure 1-1. Location of the Savannah River Site.

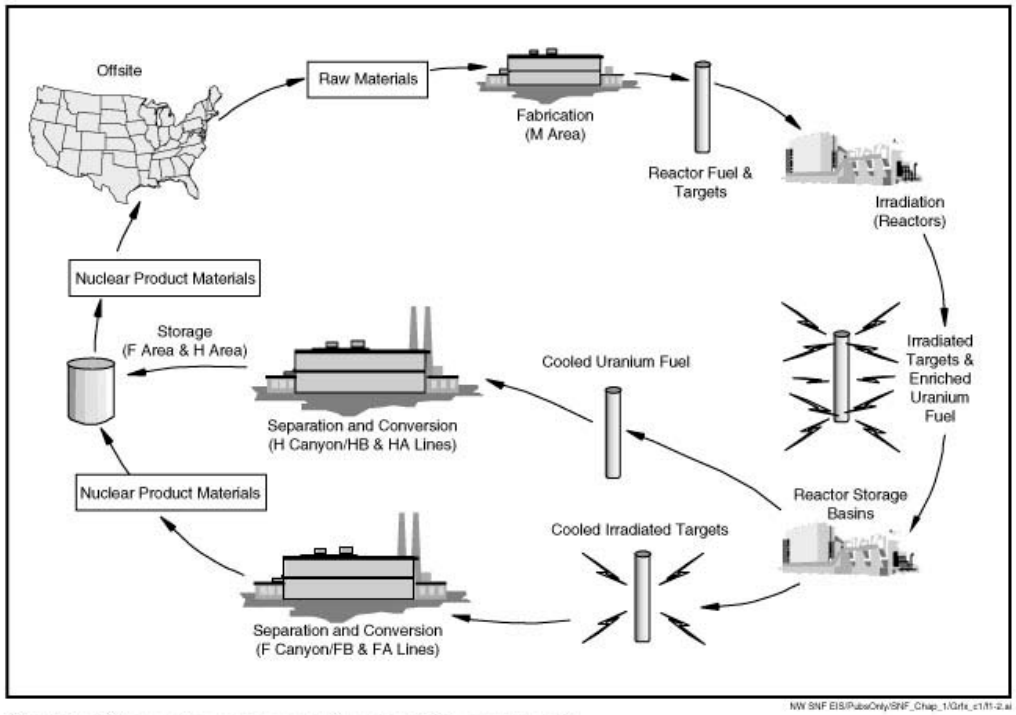

Figure 1-2. Historic nuclear materials production cycle at the Savannah River Site.

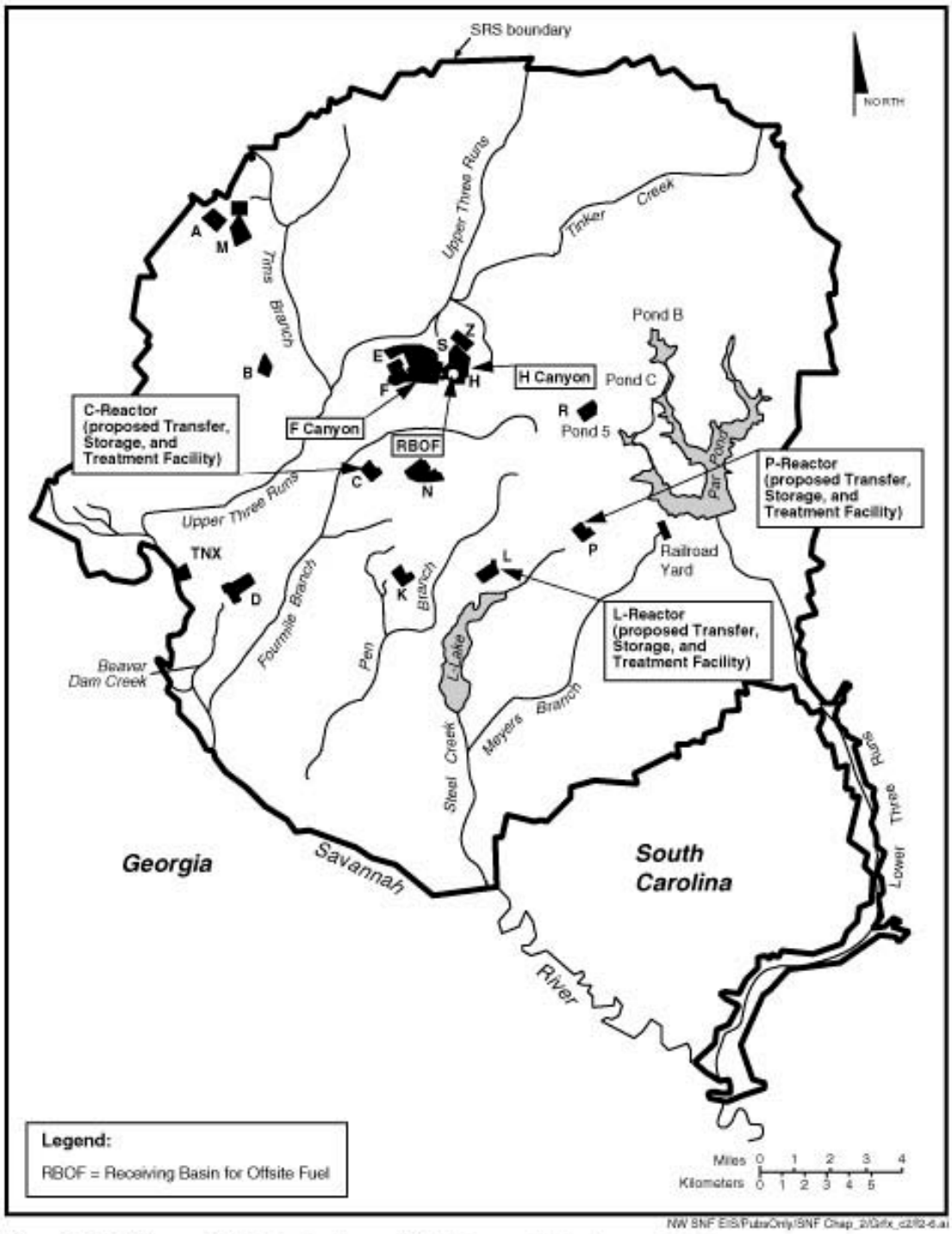

Figure 2-6. SRS map indicating locations of facilities needed for Proposed Action.

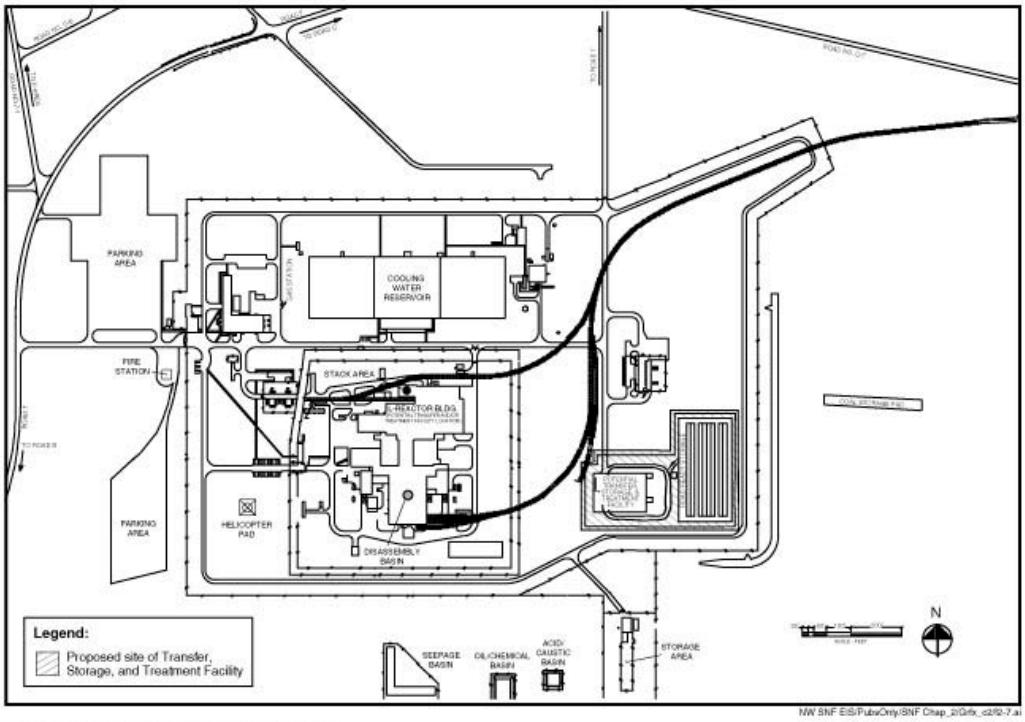

Figure 2-7. Plan view of the L-Reactor facility.

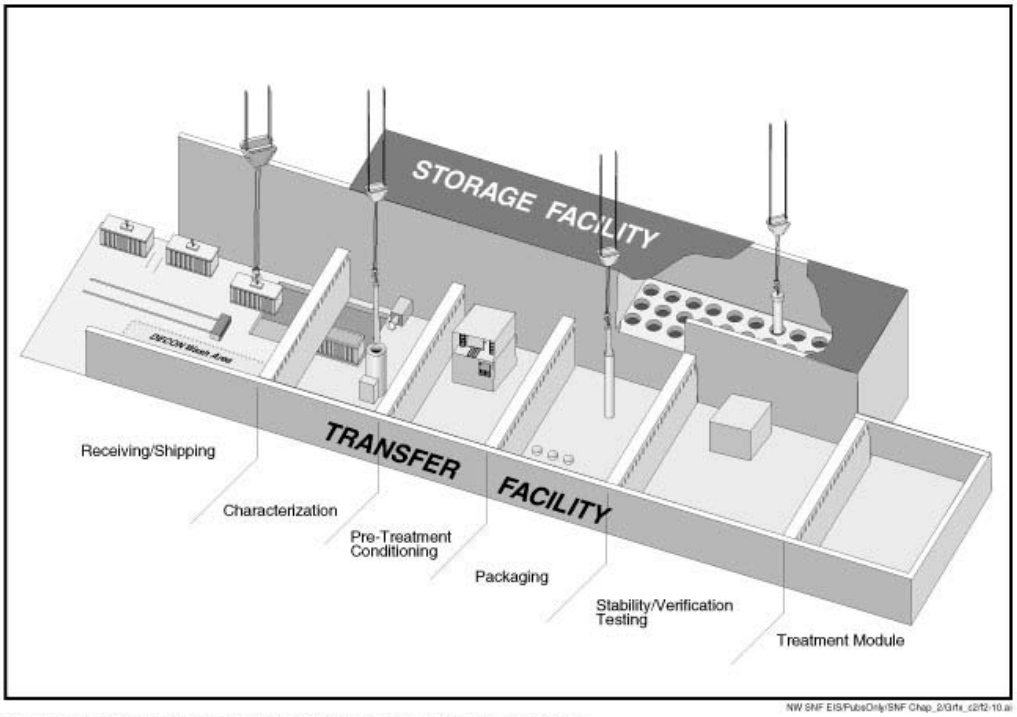

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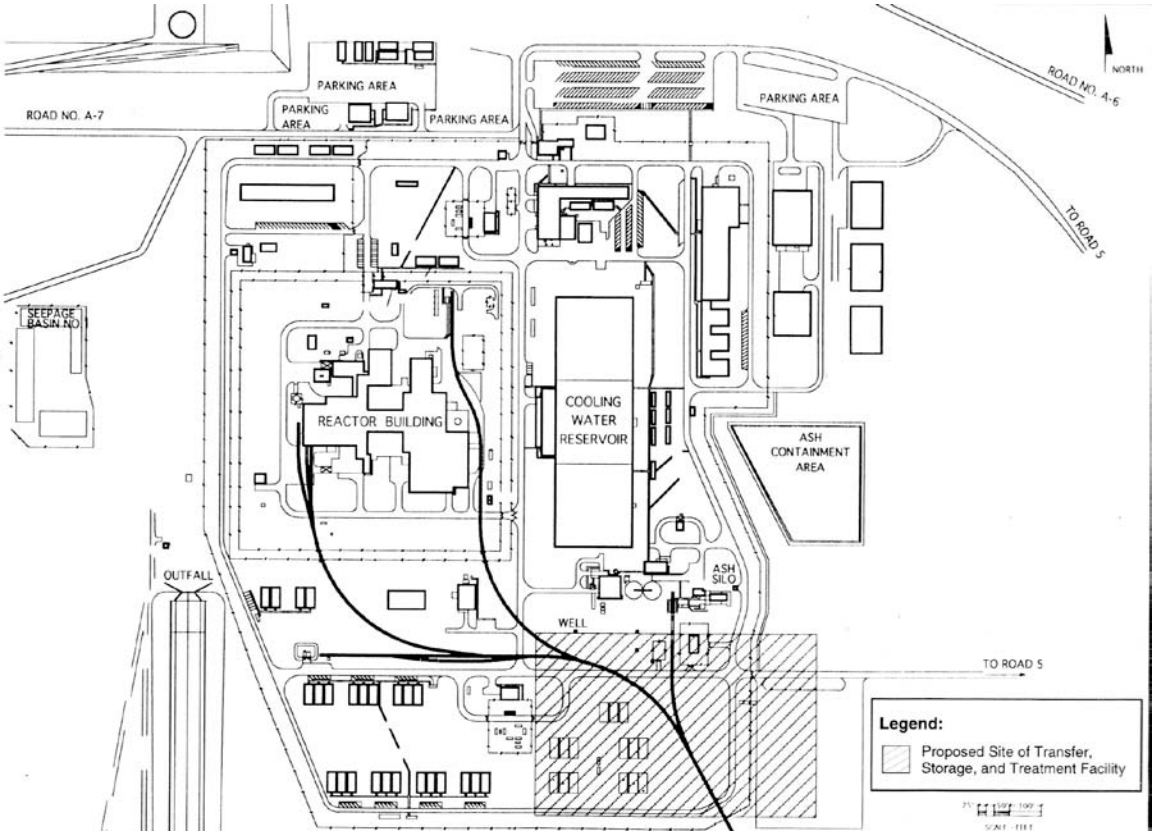
Figure 2-10. Schematic cut-a-way of the Transfer, Storage, and Treatment Facility.



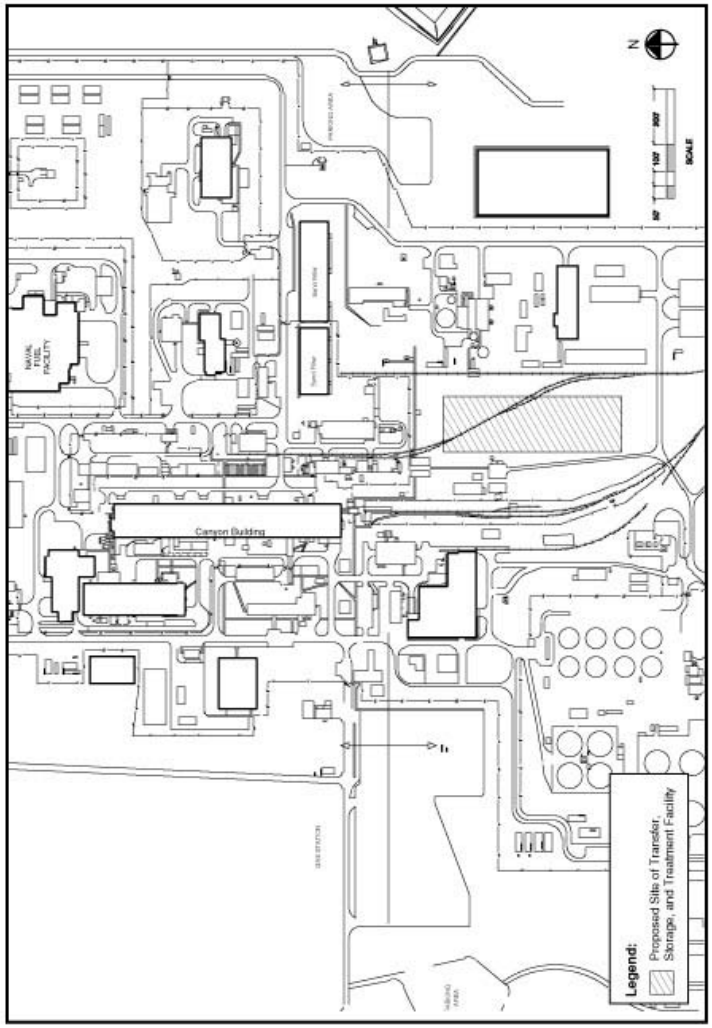


Figure 2-13. Potential Transfer, Storage, and Treatment Facility location in F Area.

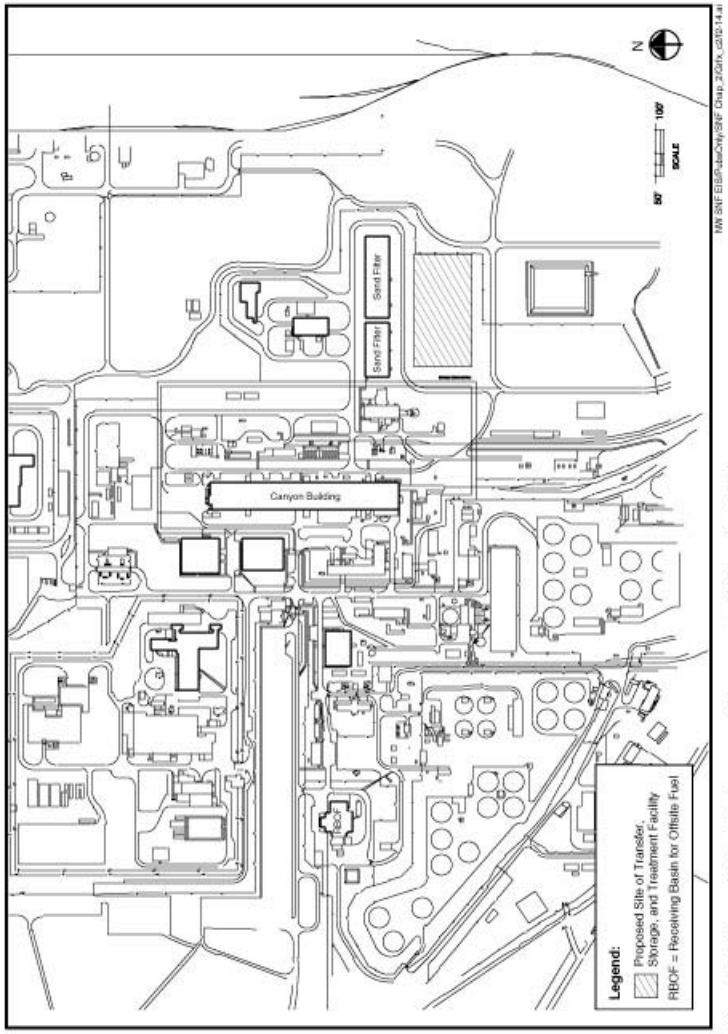
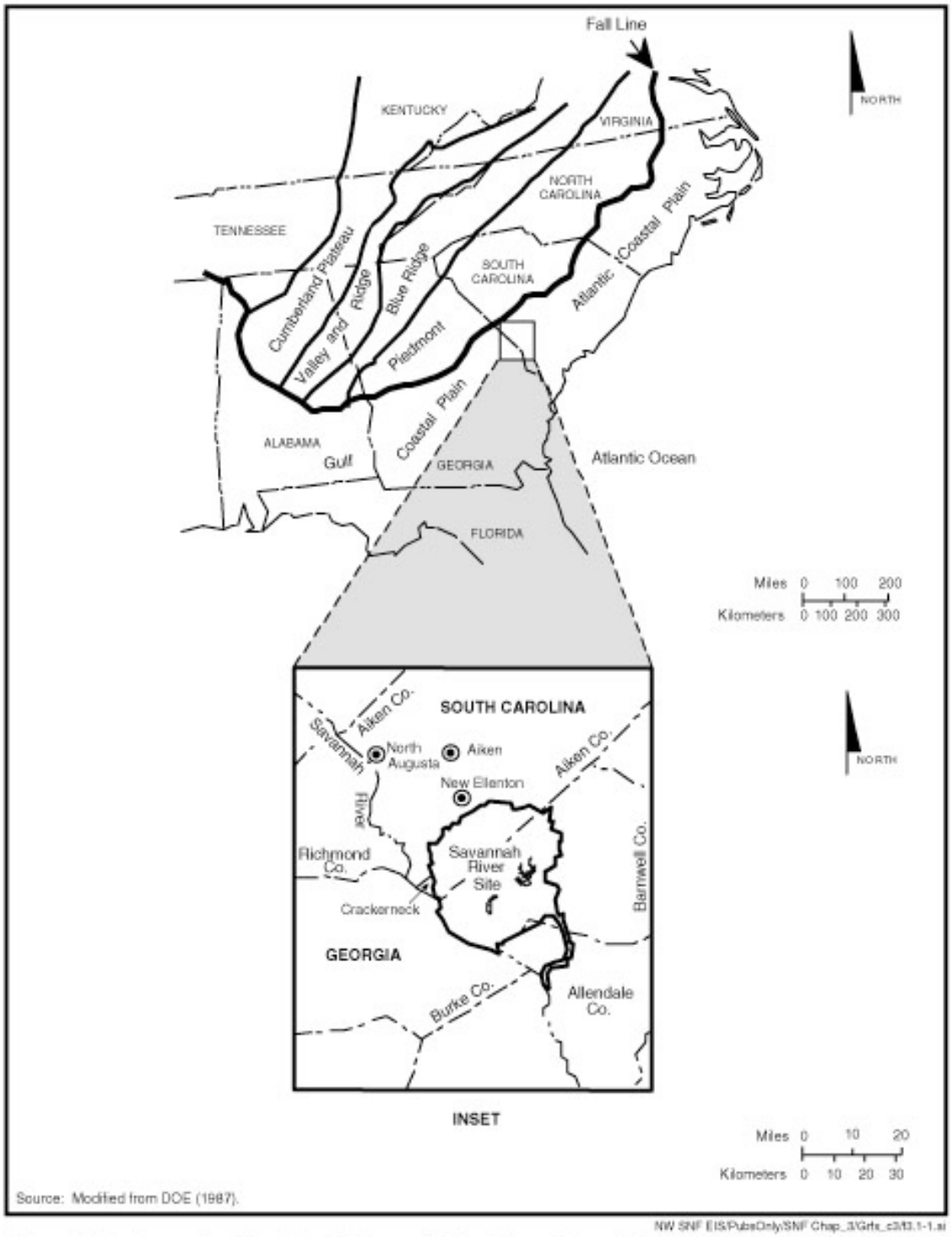


Figure 2-14. Potential Transfer, Storage, and Treatment Facility location in H Area.



Source: Modified from DOE (1987).

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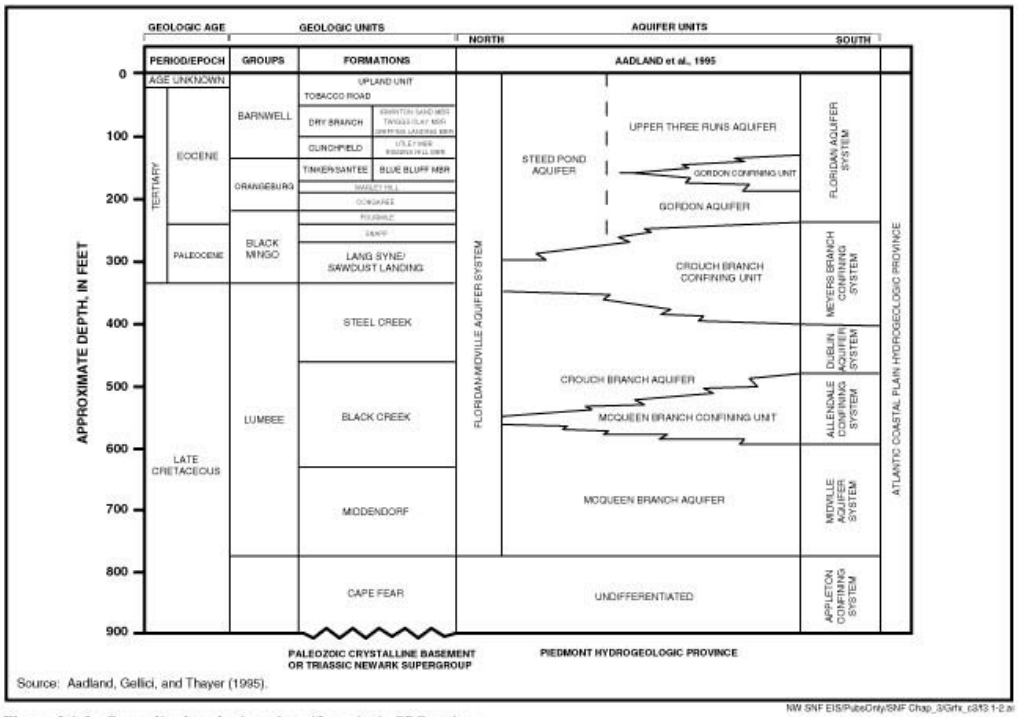
Figure 3.1-1. Generalized location of Savannah River Site and its relationship to physiographic provinces of southeastern United States.



Source: Aadland, Gellici, and Thayer (1995).

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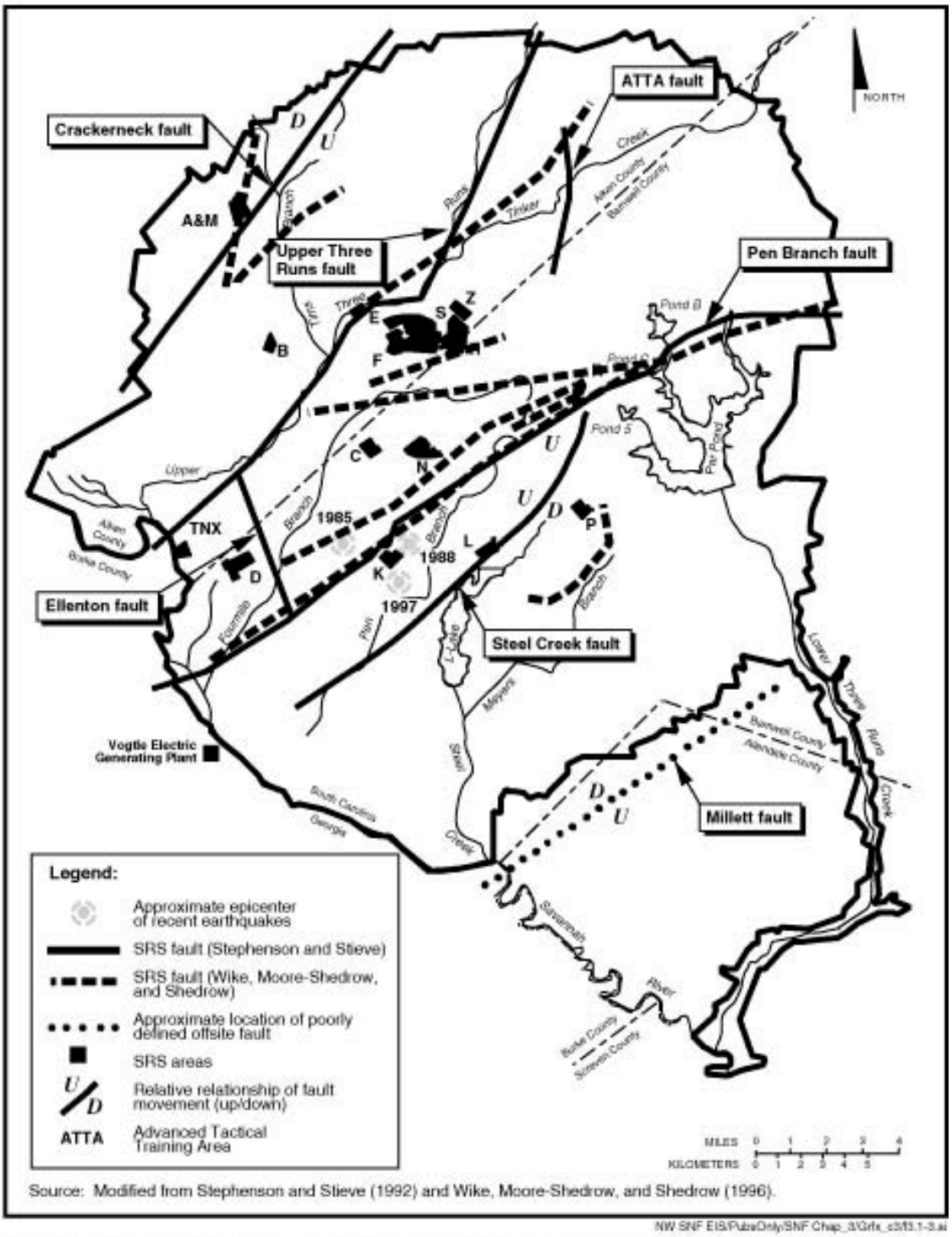
Figure 3.1-2. Generalized geologic and aquifer units in SRS region.



Source: Modified from Stephenson and Stieve (1992) and Wike, Moore-Shedrow, and Shedrow (1996).

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Figure 3.1-3. Savannah River Site, showing seismic fault lines and locations of onsite earthquakes and their year of occurrence.

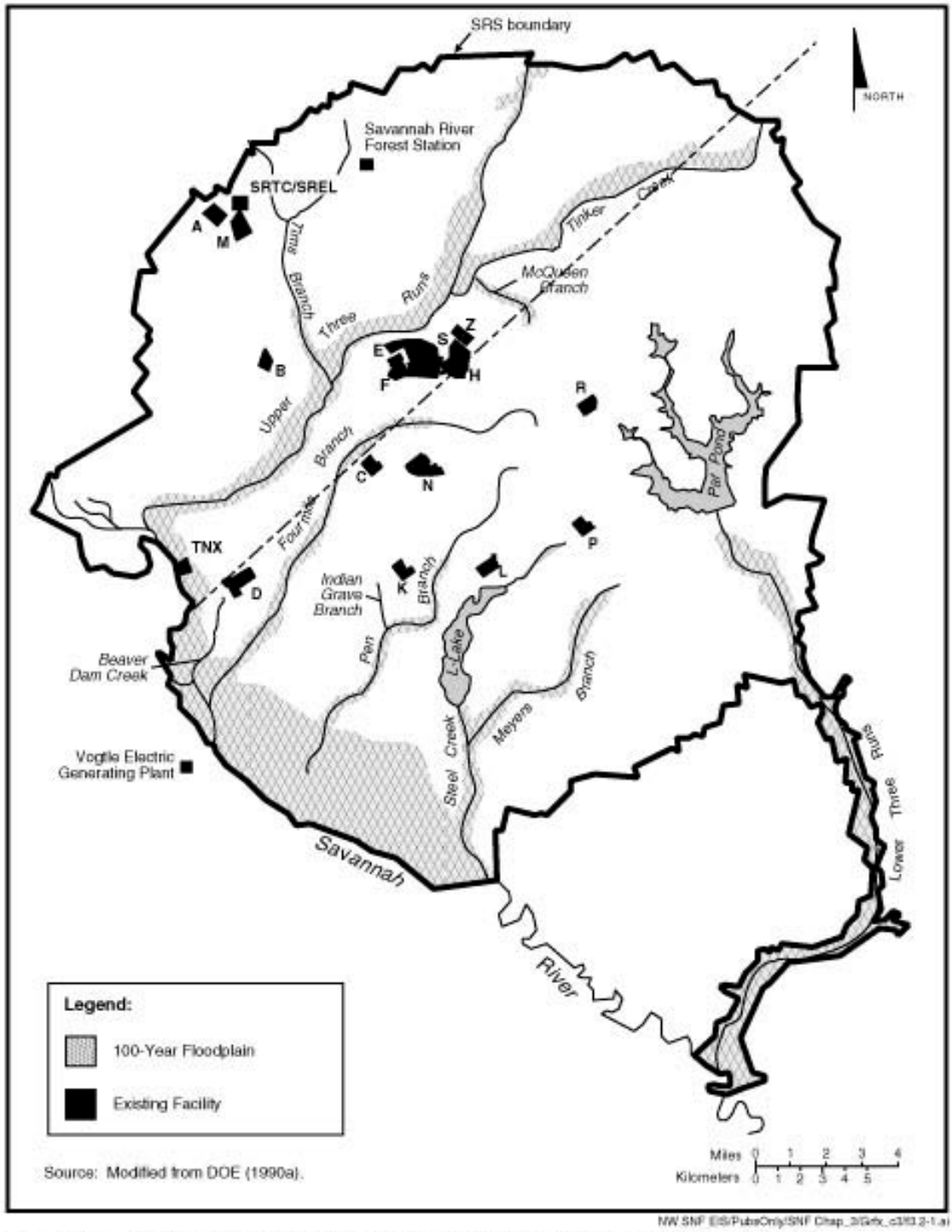

Figure 3.2-1. Savannah River Site, showing 100-year floodplain and major stream systems.

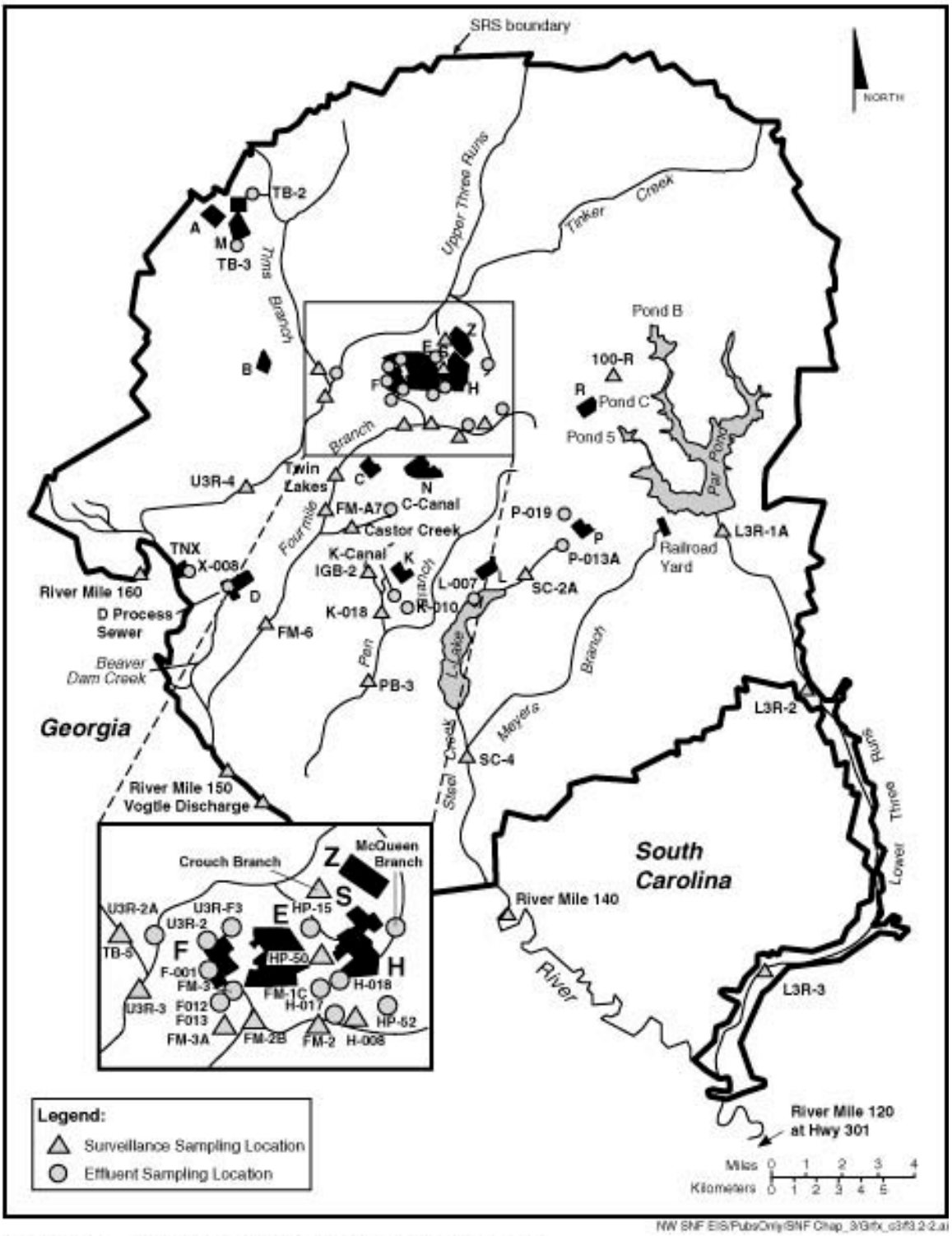

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Figure 3.2-2. Radiological surface-water sampling locations.

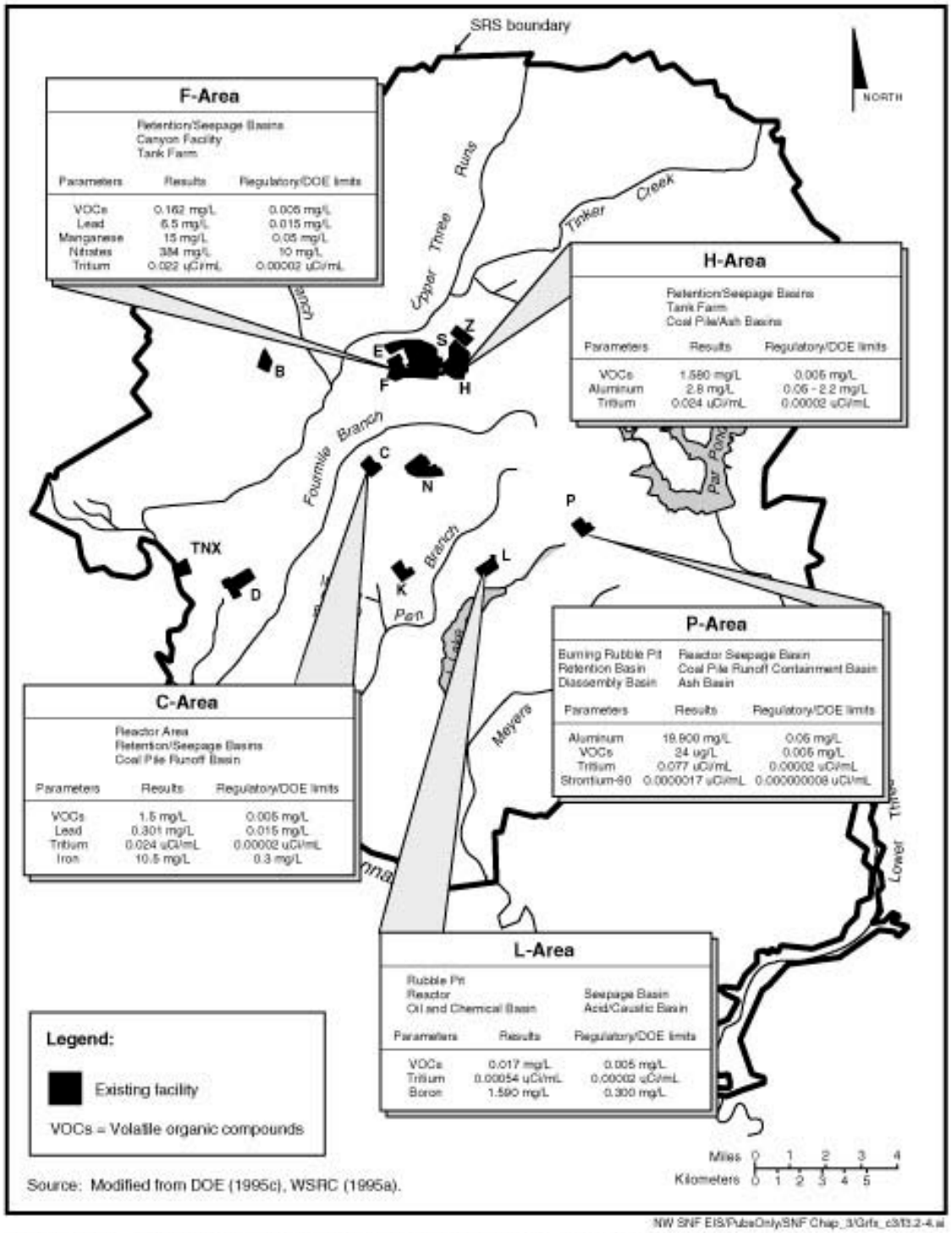

Figure 3.2-4. Maximum reported groundwater contamination at Savannah River Site.

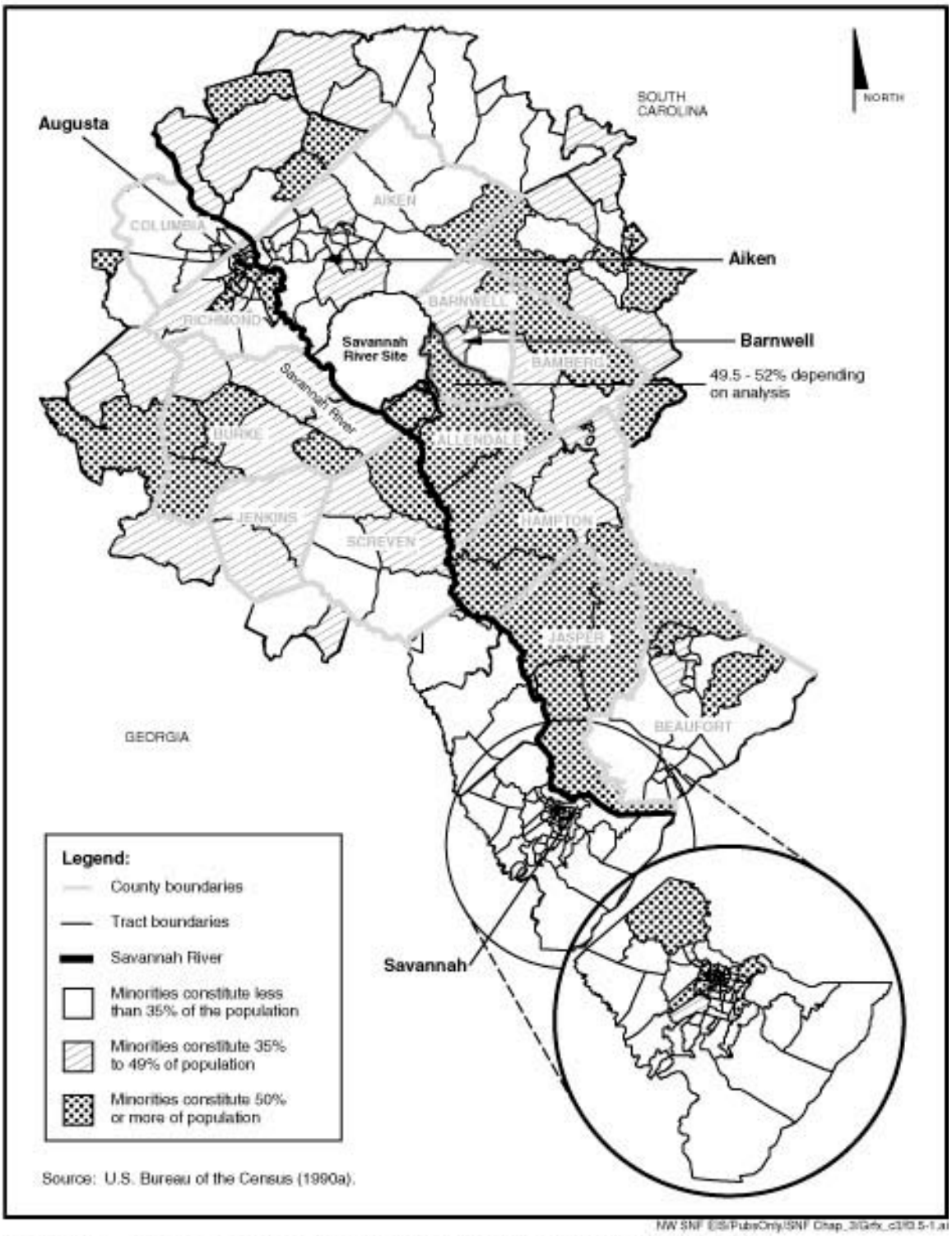

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Figure 3.5-1. Distribution of minorities by census tracts in the SRS region of analysis.

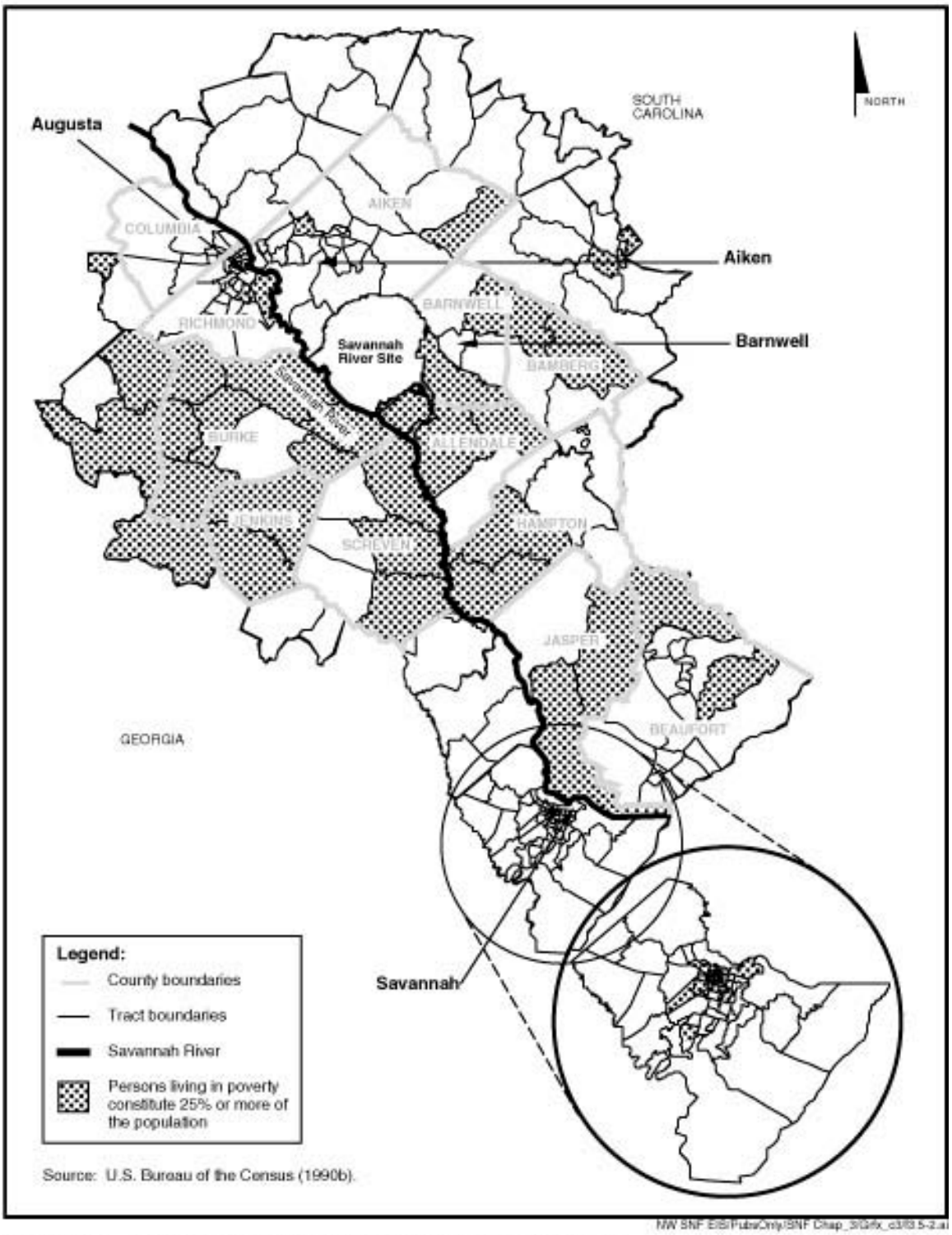

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Figure 3.5-2. Low income census tracts in the SRS region of analysis.

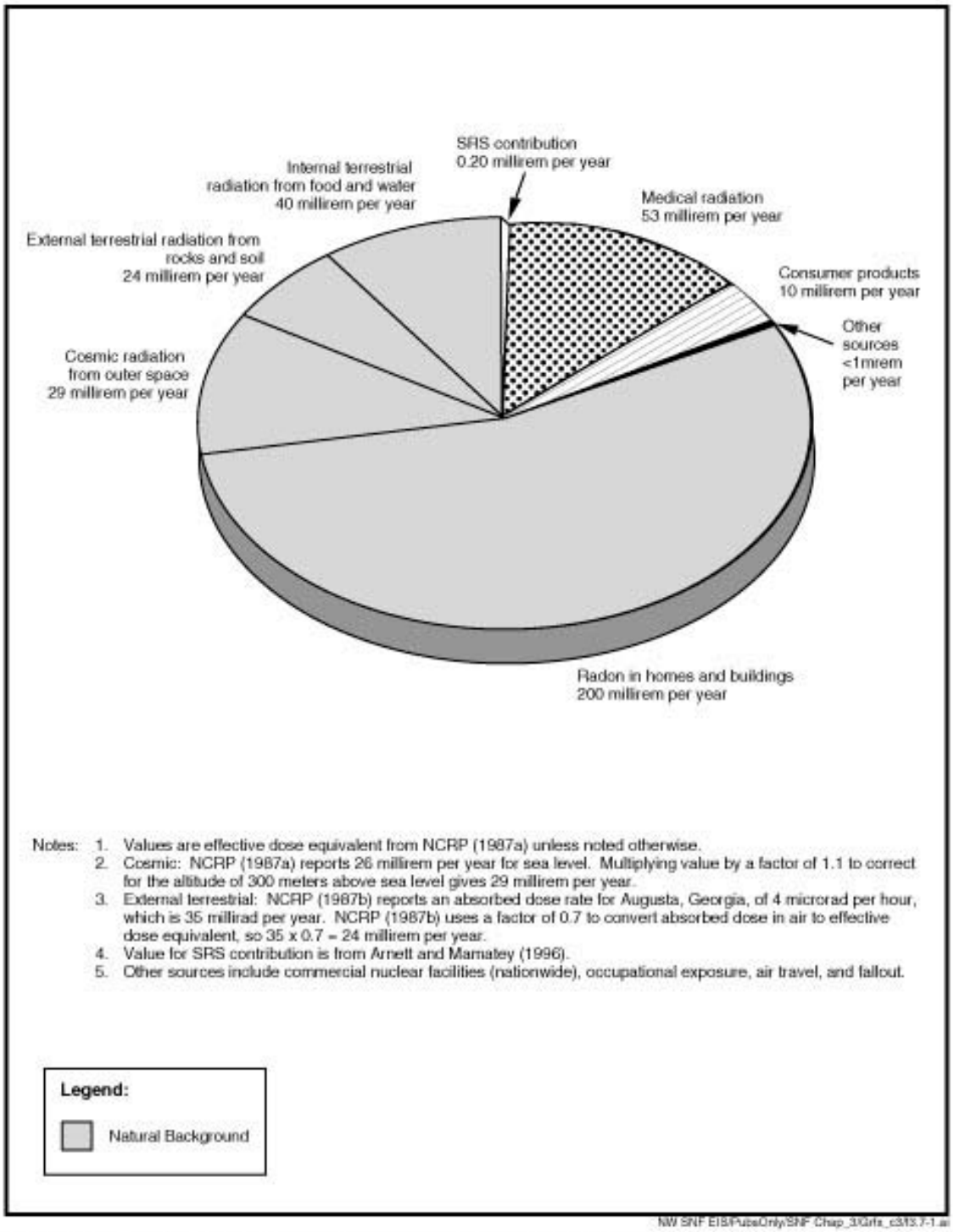

Figure 3.7-1. Major sources of radiation exposure in the vicinity of the Savannah River Site.

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Figure 4.1-8. Annular sectors around the Savannah River Site.