APPENDIX A

TECHNOLOGY DESCRIPTIONS

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APPENDIX A. TECHNOLOGY DESCRIPTIONS

This appendix describes the technology options that the U.S. Department of Energy (DOE) considered for implementing the spent nuclear fuel (SNF) management alternatives. As described in Chapter 2, DOE consolidated many of these options and eliminated two from further consideration. This appendix addresses each technology option. The options are grouped according to the technology to which they would apply.

A.1 New Packaging Technologies

A.1.1 DIRECT DISPOSAL

As the first step in the Direct Disposal process, the shipping cask would be received and unloaded in the Transfer and Storage Facility or Transfer, Storage, and Treatment Facility. The spent nuclear fuel would be placed in lag storage and the cask would be decontaminated and returned for reuse. The fuel would remain in lag storage until it was ready to be conditioned and repackaged for road-ready storage.

Conditioning activities prior to repackaging would include decanning canned fuel, cropping fuel assemblies to eliminate most of the nonfuel structural components, and limited characterization. The characterization would involve reviewing records, weighing and visually inspecting the fuel, and conducting gamma spectrometry and tests for cladding integrity. In some cases, more complete characterization could be necessary and could result in samples being taken for additional analysis. Failed fuel and other special case fuel would be recanned.

The SNF and failed fuel in cans then would be placed in canisters for road-ready storage. The fuel could be loaded into the canisters in a variety of positions, depending on the size of the fuel and its uranium content. For the Direct Disposal option, DOE could use a 24-inch (61-centimeter) diameter canister available in 5-, 10-, or 16-foot (1.5-, 3-, or 4.9-meter) lengths. Metal frames (called baskets) would be inserted into the canister to hold the fuel in a fixed position. For materials test reactor-like fuels (Group B), which make up about 97 percent of the volume of the aluminum SNF inventory, each basket would hold 16 fuel assemblies; a 10-foot (3-meter) canister could hold four baskets vertically stacked. Therefore, one 10-foot (3-meter) long canister would hold up to 64 materials test reactor-like fuel assemblies. The amount of fissile mass in the fuel could limit the length of the canister and decrease the number of fuel assemblies loaded into each canister. Dry storage space for about 1,100 24-inch (61-centimeter) diameter canisters would be needed for the Direct Disposal technology.

Before sealing the canisters, the assemblies would be vacuum-dried to remove free water. Water could contribute to continued corrosion of the fuel and to the buildup of hydrogen gas which can be generated by radiolytic decomposition of the water and by metal corrosion. Group A fuels, which are uranium or thorium metal, are more reactive than other fuels and would need more extensive drying to remove the bound water. Hot vacuum drying has been effective in eliminating bound water. However, including hot vacuum drying capability in the Transfer and Storage Facility for the small amount of Group A material (approximately 4 cubic yards [3 cubic meters]) would require a large expenditure that could be disproportionate to the benefit.

Depending on the design of the Transfer and Storage Facility, the canisters could be placed in storage singly, in storage overpacks singly or in groups, or in shipping casks (for storage) singly or in groups. Regardless of design, the fuel would be considered road-ready because no further characterization, conditioning, or other handling would be necessary before shipment. The canisters could require packaging into shipping casks, and they could need venting (to relieve buildup of hydrogen) before shipping.

Approximately 70 percent of the volume of the aluminum-based SNF to be managed at SRS would be highly enriched uranium, which would present special criticality considerations. In addition, most of the fuel considered in this EIS is aluminum-based and thus subject to more rapid corrosion (and loss of the spacing that keeps the uranium in the fuel from undergoing inadvertent criticality) than the more robust commercial or naval fuels. Finally, research reactor fuel generally has experienced lower burnup than commercial fuel, providing greater potential nuclear reactivity. Therefore, DOE proposes to address criticality by (1) a conservative limitation on the amount of fissile material in the waste package, (2) use of neutron absorbers in the fuel baskets to poison the fission chain reaction, and (3) basket and canister design to maintain subcritical geometries. For planning purposes, DOE currently limits the fissile material content to 31.8 pounds (14.4 kilograms) of highly enriched uranium per canister (DOE 1996). This limitation is based on conservative assumptions to meet current Nuclear Regulatory Commission (NRC) requirements for geologic disposal (10 CFR 60.131). DOE believes there is a good technical basis for increasing the fissile material allowance for the canisters and might do so if regulations change. Figure A-1 shows the Direct Disposal process flow.

A.1.2 DIRECT CO-DISPOSAL

From the SRS perspective, Direct Co-Disposal is identical to Direct Disposal, except 17-inch (43-centimeter) by 10-foot (3-meter) canisters would be used. The canisters would be shipped to the repository in shipping casks and repackaged into repository packages in the space among five 24-inch (61-centimeter) by 10 feet (3-meter) high-level waste canisters (see text box on page 2-5). The benefit of Direct Co-Disposal over Direct Disposal would be that little additional repository space would be needed. Because of the smaller diameter canister, approximately 1,400 dry storage spaces would be needed at the SRS for the Direct Co-Disposal technology.

A.2 New Processing Technologies

A.2.1 MELT AND DILUTE

With the Melt and Dilute technology, the shipping cask would be received and unloaded and SNF would be characterized and stored as described for Direct Disposal.

The fuel assemblies would be placed in an induction-heated melter with additions of depleted uranium and aluminum as needed to meet fissile enrichment and alloy composition specifications (see text box-Control of Melt and Dilute Process and Product Characteristics on page A-4). The melt would be contained in a crucible within the melter. The molten metal would be sampled to determine uranium-235 content and alloy composition. Adjustments to the uranium enrichment or alloy composition then could be made.

The adjusted melt would be cast into a form of approximately 16 inches in diameter and 33 inches maximum length. After cooling, the fuel would be loaded into baskets, then loaded into the canisters. The canisters would be evacuated, filled with an inert gas, sealed by welding, and transferred to road-ready storage.

About 400 canisters of the Melt and Dilute product would be produced for dry storage to be loaded as one per co-disposal package for repository disposal.

The Melt and Dilute process is a simpler technology than many of the others, especially for metal fuels. An offgas system would capture the volatile and semivolatile fission products. Oxide and silicide fuels would be reduced to metal during the melting process.

Figure A-2 shows the Melt and Dilute process flow diagram.

Figure A-1. Direct Disposal/Direct Co-Disposal process flow diagram.

Figure A-2. Melt and Dilute technology process flow diagram.

Control of Melt and Dilute Process and Product Characteristics

The Melt and Dilute treatment allows adjustment of product composition and uranium-235 enrichment to meet process and disposal requirements. Melter temperatures below about 1,000ºC, needed to maintain control of crucible interactions and offgas volumes, depend on uranium-aluminum contents of the melt. A candidate alloy composition (13.2 wt percent uranium) melts at 646ºC, with melter temperature in the range 750 to 850ºC projected for representative operations. Dilution of uranium-235 from original concentrations of as high as 93 percent down to 20 percent by addition of depleted uranium renders the melt product unsuitable for weapons use and reduces its nuclear criticality potential; lower enrichments (typically 2 to 5 percent uranium-235) further reduce criticality to the equivalent of commercial SNF.

Increased uranium content due to the addition of depleted uranium is offset by aluminum additions to maintain low melter temperatures. Dilution to 20 percent uranium-235 requires relatively small depleted uranium and aluminum additions, but dilution to lower enrichment levels requires significantly greater depleted uranium and compensating aluminum additions.

Volume increases of the final melt product due to the depleted uranium and aluminum additions result in larger numbers of waste canisters for disposal. For a product composition of 13.2 wt percent uranium, product volume is affected by final uranium-235 enrichment levels as follows:

Enrichment level, percent U-235	Number of waste canisters[a]
20	400
5	1,234
2	1,796

a. For representative inventory of processed aluminum-SNF assemblies, assuming 0.276 m^3 melt product per canister.

At the candidate alloy composition, the melt solidifies to a uniform, relatively stable microstructure of uranium-aluminum phases. Although more reactive in aqueous environments than commercial uranium oxide fuels or high-level waste glass, the melt product is well suited to characterization of reactions with waste package and geologic materials important for long-term projections of waste form behavior in a geologic repository

A.2.2 PRESS AND DILUTE

With the Press and Dilute technology, the shipping cask would be received and unloaded in the Transfer, Storage and Treatment Facility, and the SNF would undergo a limited characterization involving records review, visual inspection, and gamma spectrometry. In some cases, more complete characterization could be necessary and could result in samples being taken for additional analysis elsewhere. The characterization data would be used to determine the amount of depleted uranium needed to meet dilution requirements (if any).

The fuel assemblies would be cropped to eliminate most of the nonfuel structural components and reduce storage space. The fuel assemblies would be vacuum-dried to remove free water.

The dried assemblies would be placed in a mechanical press for compaction. The pressed spent nuclear fuel would be layered with depleted uranium and pressed again to lock the pieces together. Layering would continue to the limits imposed by the canister dimensions. The shape of the pressed assembly would be determined by future research but would be optimized to reduce free space in the canister. Free space could result in the intrusion of a moderator (e.g., water), thereby changing the assumptions under which nuclear safety calculations were performed. The final shape of the waste form could be cylindrical (from molds) or stacked disks.

Finally, the pressed fuel form would be placed into 17-inch (43-centimeter) diameter canisters, which would be filled with an inert gas and welded closed. The pressing operation and the canister loading operation would be controlled to limit the fissile material in the canisters, in accordance with nuclear criticality considerations. The number of canisters produced for dry storage would be about 630 to be loaded as one per co-disposal overpack for repository disposal.

The primary advantage of Press and Dilute technology is its simplicity. However, the variable sizes of Group C SNF might make the technology unsuitable for those fuels without special disassembly before compaction. Particulate fuels (Group D) would not be amenable to pressing. Figure A-3 shows the Press and Dilute process flow diagram.

A.2.3 CHOP AND DILUTE

In the Chop and Dilute treatment, the shipping cask would be received and unloaded, and the SNF would undergo a limited characterization as described in Section A.2.2 Press and Dilute. The fuel assemblies would be cropped to eliminate most of the nonfuel structural components and reduce storage space. The assemblies would be vacuum-dried to remove free water.

The dried assemblies would be fed into a shredder. Similarly shredded depleted uranium-aluminum alloy would be combined with the shredded fuel to produce a mix of reduced enrichment. The shredded fuel would be placed into 17-inch (43-centimeter) diameter canisters, which would be filled with inert gas and welded closed. The canister loading would be controlled to limit the amount of fissile material in the canisters in accordance with nuclear criticality requirements. The number of canisters produced for dry storage and repository disposal would be about the same as for the Press and Dilute process (630).

The material resulting from Chop and Dilute would not be homogeneous and would result in a considerable amount of free space in each canister. The free space would contribute to an increase in the number of canisters required and could increase vulnerability to a nuclear criticality. In addition, the material could be pyrophoric. Because of these difficulties with Chop and Dilute, DOE considers it to be the least attractive of the three dilution technologies (Melt and Dilute, Press and Dilute, and Chop and Dilute).

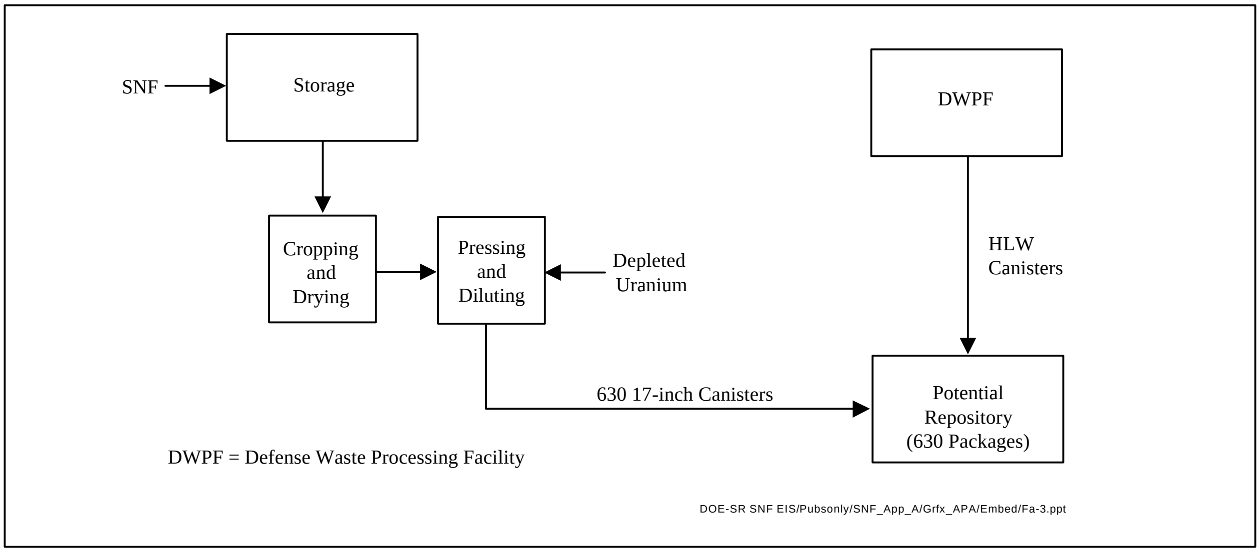

Figure A-3. Press and Dilute process flow diagram.

A.2.4 DISSOLVE AND VITRIFY

The SNF would be cropped and charged to an electrolytical dissolver similar to that used in H Canyon. The electrolyte solution would be nitric acid saturated with boric acid. The process would operate in a batch mode to ensure criticality control. Depleted uranium would be added, as needed, to reduce the uranium-235 enrichment to approximately 5 percent.

The dissolver solution would be transferred to a holding tank for chemical and radiological analyses to determine the need for any adjustments prior to the vitrification step. The solution then would be transferred to an electrically-heated melter, along with glass-forming chemicals. Several dissolver batches could be melted at once. The resulting molten glass, having been preanalyzed in the holding tank, should be of sufficient quality to be poured into canisters similar to those used at the Defense Waste Processing Facility. About 1,350 canisters would be produced for emplacement in about 270 repository waste packages.

Although DOE could perform dissolution using the existing equipment at H Canyon, the analysis in this EIS assumes the construction of a new Dissolve and Vitrify facility. Figure A-4 shows the Dissolve and Vitrify process flow diagram.

A.2.5 GLASS MATERIAL OXIDATION AND DISSOLUTION SYSTEM

The Glass Material Oxidation and Dissolution System (GMODS) converts SNF directly to borosilicate glass using a batch process. Criticality concerns are addressed by diluting the uranium-235 enrichment with depleted uranium and using boron oxide as a dissolving agent (boron is a neutron poison). Although the addition of depleted uranium and glass frit adds to the mass, the high-density, monolithic glass still would provide a smaller volume for dry storage than would Direct Co-Disposal.

The principal piece of equipment for GMODS would be an induction-heated cold-wall melter, which in commercial use converts corrosive or high-melting metals to ultrapure materials. The melter would be charged with a molten glass consisting of lead oxide and boron oxide. The lead oxide converts the metals in the SNF to oxides; oxides and amorphous materials tend to dissolve in molten glass, but metals do not. Boron oxide is a common agent for dissolving oxides into glass (e.g., welding slag). A problem

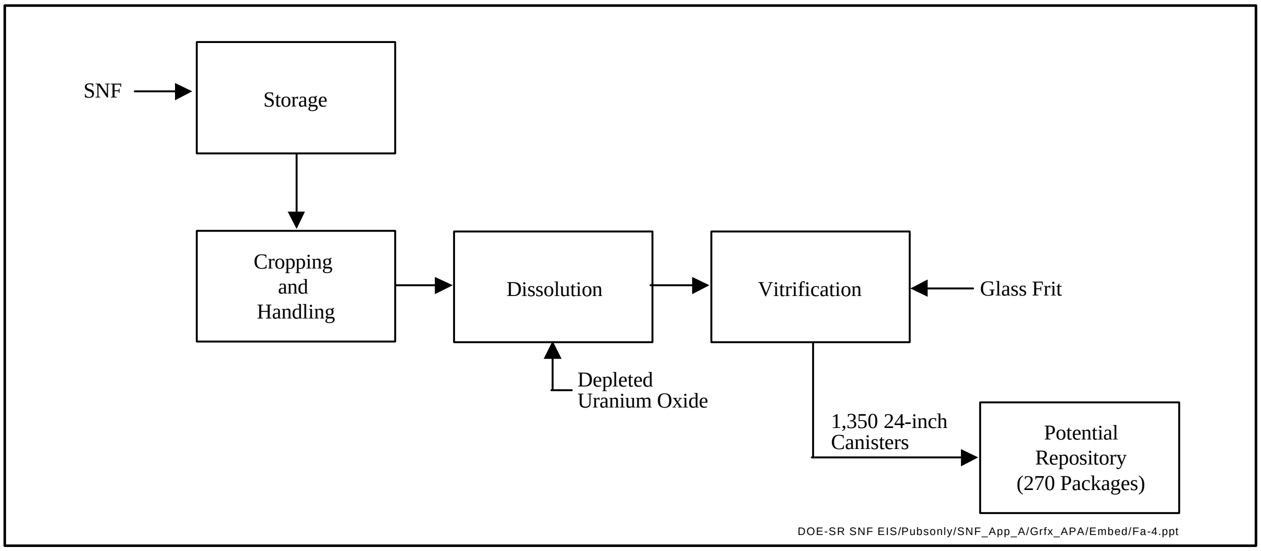

Figure A-4. Dissolve and Vitrify process flow diagram.

with using lead oxide is its corrosivity, which could affect the service life of the melter.

As the SNF is fed into the melter, the aluminum, uranium, and other metals would be oxidized and dissolved in the molten glass. Uranium oxides and other oxides would be directly dissolved. The oxidation of the metals converts the lead oxide to metallic lead, which sinks to the bottom of the melter.

The resulting glass mixture would not have qualities necessary for long-term durability, so silicon oxide (glass frit) additions would be necessary to increase the durability. The silicon oxide would not be part of the initial melter charge because its properties are not conducive to rapid oxidation-dissolution of SNF. Unreduced lead oxide could limit the durability of the glass, and increase volume, so carbon would be added to the melt to reduce the excess lead oxide.

The glass melt would be decanted from the melter and formed into glass marbles. For criticality and other practical reasons, the batch melts using GMODS would not be large enough to fill a 24-inch (61-centimeter) diameter canister. Therefore, the glass marbles would be stored and remelted, allowing a continuous pour to fill several 24-inch diameter canisters at a time. The GMODS process would produce typically about 1,350 canisters for emplacement in about 270 repository packages.

After decanting the glass, the melter would be recharged with boron oxide and, if necessary, lead oxide. Oxygen would be piped into the system to convert the metallic lead at the bottom of the melter back to lead oxide. Lead would be an oxygen carrier that did not leave the system.

Radioactive offgases produced during this process would be filtered and treated appropriately. Figure A-5 shows the Glass Material Oxidation and Dissolution process flow diagram.

A.2.6 PLASMA ARC TREATMENT

The Plasma Arc Treatment technology uses a plasma torch to melt and oxidize the SNF in conjunction with depleted uranium oxide and other ceramic-forming materials as necessary. The fuel would be fed into the process with minimal sizing or pretreatment. The plasma arc would cut the fuel assemblies into small pieces and heat the fuel to temperatures as high as

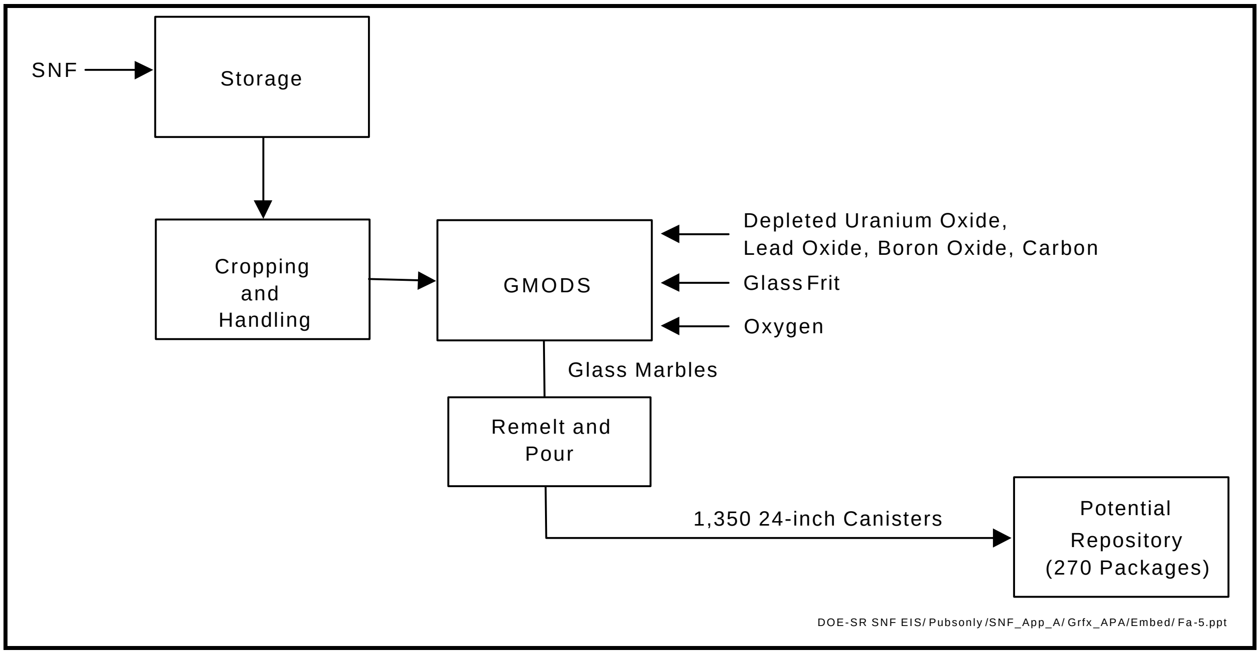

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Figure A-5. Glass Material Oxidation and Dissolution process flow diagram.

1,600°C (2,900°F) to melt and oxidize it in a rotating furnace. Ceramic material would be added as necessary with the mixture homogenized by the torch. When melting and oxidation were complete, the rotating furnace would slow and the melt would fall into molds prepared to receive it.

Some types of SNF might not require the addition of ceramic material to the process because the oxidation would produce a robust ceramic form from the fuel itself. Many metallic fuels would, however, need some ceramic addition. Depleted uranium could be added to the process in almost any form to reduce the uranium-235 enrichment. Criticality issues would be addressed by limiting the process to batch runs of preselected quantities of fissile material, by the addition of the depleted uranium, and by the addition of neutron poisons if necessary. The Plasma Arc treatment would produce about 490 canisters to be contained in 98 repository packages.

As with all processes that dissolve or melt the SNF, the Plasma Arc Treatment would produce radioactive offgases. These gases would be filtered and treated by appropriate means, with the filter and treatment media recycled into the plasma arc furnace for incorporation into the ceramic product. Figure A-6 shows the Plasma Arc Treatment process flow diagram.

A.2.7 ELECTROMETALLURGICAL TREATMENT

The Electrometallurgical Treatment process would adapt a technology under development at the Argonne National Laboratory for processing Experimental Breeder Reactor-II fuel and blanket assemblies. The process has been demonstrated for the stainless steel-clad uranium alloy fuels used in this reactor. The electrorefining process employs a technology used in industry to produce pure metals from impure metal feedstock. The feasibility of the Electrometallurgical Treatment for aluminum-based fuels has been tested in the laboratory and is theoretically possible as conducted in the following two stages. An electrorefiner facility is available at the Idaho National Engineering and Environmental Laboratory to test development concepts.

Preparation

Before electrorefining, the fuel would be cropped and the end fittings discarded. The fuel assemblies would be compacted and melted with

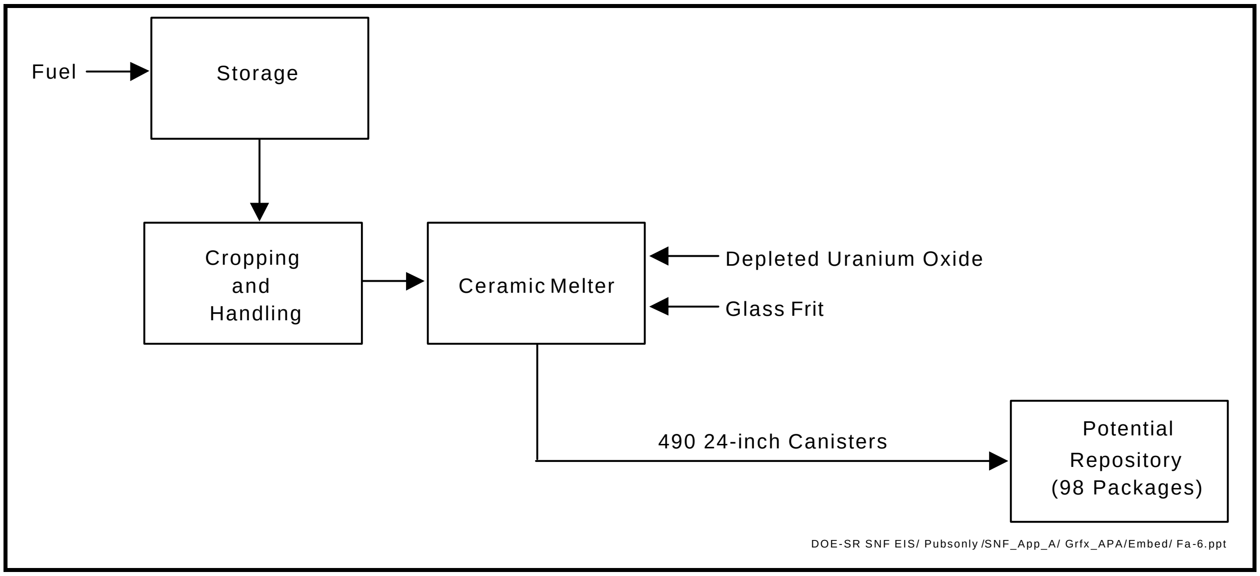

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Figure A-6. Plasma Arc Treatment process flow diagram.

silicon added to the melt to complex the uranium and enhance its separation from aluminum in the subsequent electrorefining step.

Melting would vaporize many volatile fission products (e.g., cesium, rubidium, bromine, iodine, xenon, and krypton), which, with the exception of the noble gases, would be captured in a fibrous aluminosilicate trap. The molten fuel would be poured into ingots that would become the anodes for processing in the next step in the electrorefiner.

Aluminum Separation

The electrorefining process would first use a lithium fluoride-potassium fluoride electrolyte to separate the aluminum from the anode. Aluminum and alkaline earth fission products would dissolve out of the ingot; the aluminum would form a soluble compound of potassium aluminum hexafluoride (K_3AlF_6), which would travel to the cathode where it would be reduced to pure aluminum metal. The alkaline earth fission products would remain in the electrolyte. The aluminum deposits on the cathode would be continually scraped off and collected.

Because some electrolyte salts would be entrained with the aluminum, the aluminum would be melted to separate the aluminum from the salts. The melt would cool below the melting point of the aluminum, and the salts would be poured off and recycled. The aluminum would be disposed as low-level waste.

Uranium Separation

After essentially all aluminum was removed from the anode, the actinides (primarily uranium), rare earths, and noble metals would remain. The anode would be placed in a second refiner that used lithium fluoride, potassium fluoride, and uranium trifluoride salts as electrolytes. The uranium in the anode present as a uranium silicide would be oxidized to uranium trifluoride and transported to the cathode where it would be reduced to uranium metal. The uranium metal deposits would be collected and separated from electrolyte salts as was the aluminum.

Salt Scrubbing

With the continued electrorefining of the SNF, alkaline earth fluorides would build up in the aluminum electrorefiner, and the rare earth and transuranic fluorides would build up in the uranium electrorefiner. These waste products could be separated from the electrolyte by ion exchange or chemical reduction and oxide precipitation.

Waste Treatment

The Electrometallurgical Treatment process would produce several waste streams besides aluminum: scrubbed alkaline earths, rare earths, transuranics, the metal remaining in the anode after uranium electrorefining, and the fibrous aluminosilicate filter used to collect volatile fission products released during SNF melting. These wastes would be placed in an air oxidation furnace to burn to an oxide powder with noble metal fines dispersed in the powder. A small glass melter would melt the oxide powder with glass-forming materials to produce a glass similar to that produced in the Defense Waste Processing Facility. The glass would be formed into marbles for shipment to the Defense Waste Processing Facility for incorporating into high-level waste glass logs. The electrometallurgical treatment would produce about 90 24-inch diameter canisters to be contained in 18 repository disposal packages.

Uranium Dilution

A small melter would melt the uranium metal and blend it with depleted uranium to produce a uranium enriched to about 5 percent uranium-235. This uranium could be sold as feedstock for commercial nuclear fuel manufacture. Figure A-7 shows the Electrometallurgical Treatment process flow diagram.

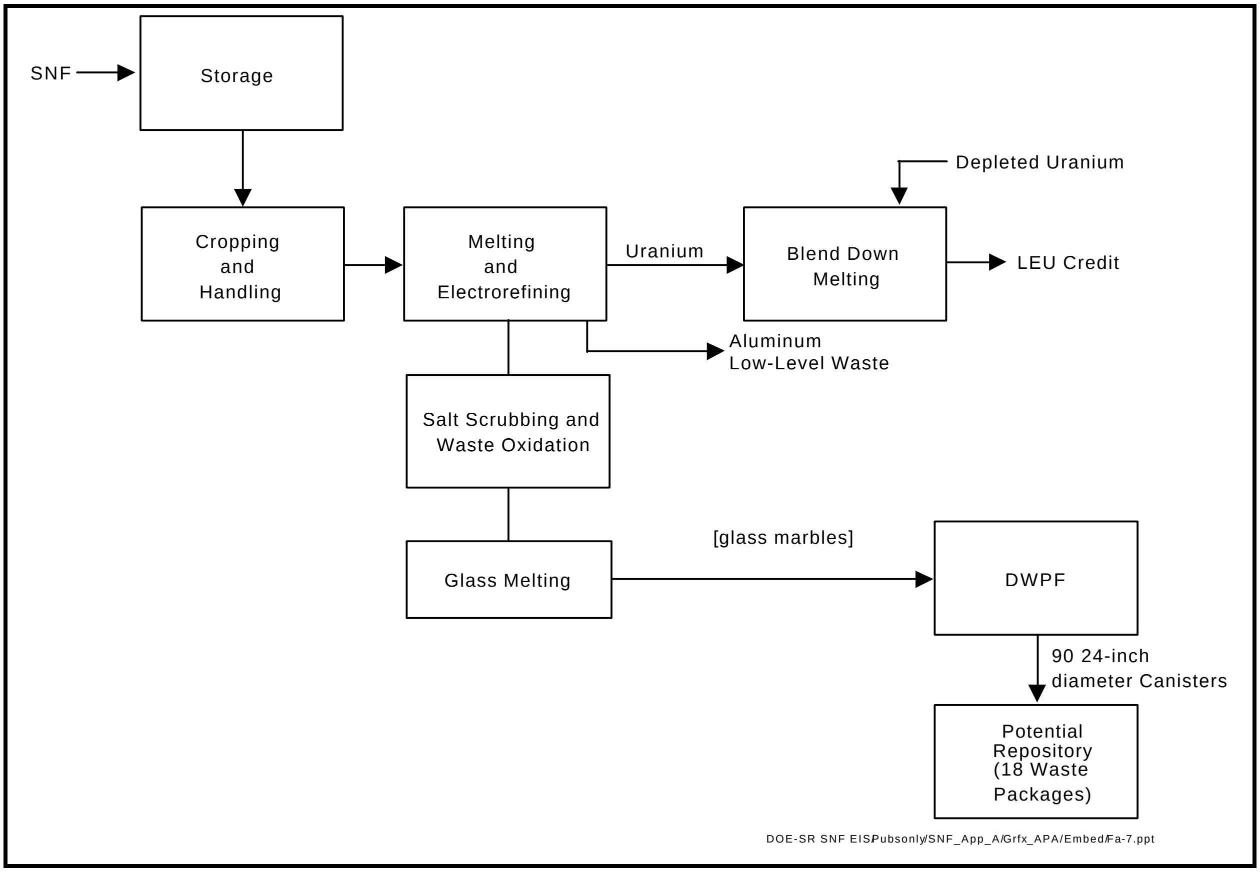

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Figure A-7. Electrometallurgical Treatment process flow diagram.

A.2.8 TECHNOLOGIES NOT ANALYZED

This section describes technologies that DOE considered but did not analyze further in the EIS because the technologies need further research to demonstrate they are technically viable and cost effective. These technologies have not undergone bench-scale demonstrations.

Chloride Volatility

Chloride volatility is an advanced treatment technology being investigated at the Idaho National Engineering and Environmental Laboratory. The process segregates major nonradiological constituents from SNF for the purpose of volume reduction, and isolates the fissile material to produce a glass or ceramic waste form.

The process is based on completely volatilizing the fuel elements and separating the gaseous constituents. The fuel would react with chlorine gas at a temperature greater than 1,200°C (2,200°F) to produce volatile chlorides. The fission products and transuranics would be separated by passing the gas through molten zinc chloride in a counter-current scrubber. The gases minus the fission products and transuranics would flow through a series of condensers to remove chloride compounds by fractional distillation. The series of uranium chlorides could be recovered separately, if desired.

The molten zinc chloride would be regenerated by vacuum distillation to recover it for recycle. The fission product and transuranic residue would be converted to oxides or fluorides by fluorination for vitrification and melting with glass frit additives. As an alternative, the residues could be oxidized by boric acid at high temperatures. The transuranics could be separated from the fission products by solvent extraction if separate disposal were necessary.

Although there has been no experimental work with this technology, DOE has determined that the time and expense to overcome the technical risks would be too great. Therefore, this technology is not analyzed further in this EIS.

Can-in-Canister

The Can-in-Canister concept was developed to dispose of excess plutonium. The method would place an array of stainless-steel cans containing plutonium ceramic in a high-level waste canister. The molten high-level waste glass would be poured around the cans. The placement structure would maintain spacing between the cans and the wall of the canister. The Can-in-Canister method is a potentially favorable method for disposing of plutonium because the radiation fields emanating from the high-level waste would discourage intrusion to recover the plutonium. Plutonium itself does not produce high radiation fields.

Can-in-Canister technology is not as attractive for SNF. Most SNF produces high radiation fields that would render recovery difficult and thus would not need the added deterrent of high-level waste surrounding it. In addition, because the melting point of aluminum is less than that of glass in high-level waste vitrification operations, the aluminum fuel could melt in the cans as the canister was being filled with the molten high-level waste glass. Finally, it is not certain the integrity of the glass could be maintained if it contained large voids - in common with the disposal of plutonium in glass.

The Direct Co-Disposal technology provides all the benefits of Can-in-Canister technology without the disadvantages. The SNF would be surrounded by high-level waste glass canisters, ensuring that HLW radiation fields would render the SNF inaccessible for long periods of time. The fuel would not displace any high-level waste canisters, thus eliminating the need for additional repository waste packages. Also, the SNF would not be heated near or over its melting point. For these reasons, the Can-in-Canister process was not analyzed in this EIS.

A.3 Conventional Processing

As discussed in Chapter 2, DOE could use F or H Canyon to process SNF. F Canyon historically has been used to recover depleted uranium and plutonium from depleted uranium target materials irradiated in SRS reactors. H Canyon historically has been used to recover highly-enriched uranium and neptunium from SNF. The following paragraphs are applicable to operations in H Canyon. F Canyon operations would be similar.

At the SNF wet storage basins, the fuel would be placed in aluminum bundle sleeves. For materials test reactor-like fuel elements, the fuel would be stacked four to five elements high in the bundle sleeve. Before shipment to H Canyon, the bundle sleeves would be assembled into larger arrays to make a complete bundle. The size of the array would be determined by shipping cask and the size of the dissolver, and by criticality concerns. Bundling would facilitate handling and maintain a noncritical geometry as the fuel was charged to the dissolver. The storage racks in the Receiving Basin for Offsite Fuel and L-Reactor Disassembly Basin use bundle sleeves to maximize storage space.

The SNF would be transported in a water-filled cask on a rail car from either L-Reactor Disassembly Basin or the Receiving Basin for Offsite Fuel. Inside the airlock doors to the hot canyon, the fuel would be unloaded and placed in an interim wet storage basin to await processing. The bundles of SNF would be fed into the top of a dissolver tank. The fuel would be dissolved in hot nitric acid, producing a solution of highly-enriched uranium, fission products, aluminum, and small amounts of transuranic materials such as neptunium and plutonium.

Head-end processing would use two clarification steps to remove undesirable contaminants that could impede the subsequent solvent extraction process. Gelatin would be added to precipitate silica and other impurities. The clarified solution would be adjusted with nitric acid and water in preparation for the first-cycle solvent extraction. The waste stream generated from the head-end process would be chemically neutralized and sent to the high-level waste tanks.

The first-cycle solvent extraction in the hot canyon would remove the fission products and other impurities, and then separate the uranium from the other actinides. Nonuranium actinides would not be recovered. If necessary, a second-cycle solvent extraction could further purify the uranium solution. The solvent would be recovered for reuse, the acid solution containing the fission products would be neutralized and transferred to the high-level waste tanks, and the uranium in a uranyl nitrate solution would be transferred to H-Area tanks to be blended down to about 5 percent uranium-235. The uranyl nitrate could be made available for commercial sale.

Chemical processing would generate liquid high-level waste, for which SRS has existing storage and treatment facilities. Impacts associated with the operation of these facilities are described in the Interim Management of Nuclear Materials (IMNM) EIS (DOE 1995) and the Defense Waste Processing Facility (DWPF) Supplemental EIS (DOE 1994). Chapter 5 summarizes the results from the IMNM and DWPF EISs. For completeness, the following paragraphs summarize high-level waste processing at SRS.

Chemical processing produces an acidic solution that is neutralized before transfer to large tanks in the F- and H-Area Tank Farms. During storage to allow short-lived radionuclides to decay, the insoluble components of the alkaline waste settle to the bottom of the tank to form a sludge layer. The liquid supernate is decanted and evaporated to concentrate it into a crystallized salt. Evaporator overheads are condensed and discharged to the F/H Effluent Treatment Facility.

In preparation for final disposal, the salt is redissolved and processed to separate it into high-radioactivity and low-radioactivity fractions. The high-radioactivity fraction is sent to the Defense Waste Processing Facility where it is incorporated into a glass form for eventual disposal in a geologic repository. The low-radioactivity fraction is sent to the Saltstone Manufacturing and Disposal Facility where it is mixed with cement, slag, and flyash to produce a cementatious grout solidified in onsite disposal vaults.

The sludge in the high-level waste tanks, after washing to remove dissolved salts, also is transferred to the Defense Waste Processing Facility for incorporation into the high-level waste glass form for repository disposal. About 150 canisters of 24-inch diameter would be produced during the conventional processing of spent nuclear fuel.

Figure A-8 shows the Conventional Processing flow diagram.

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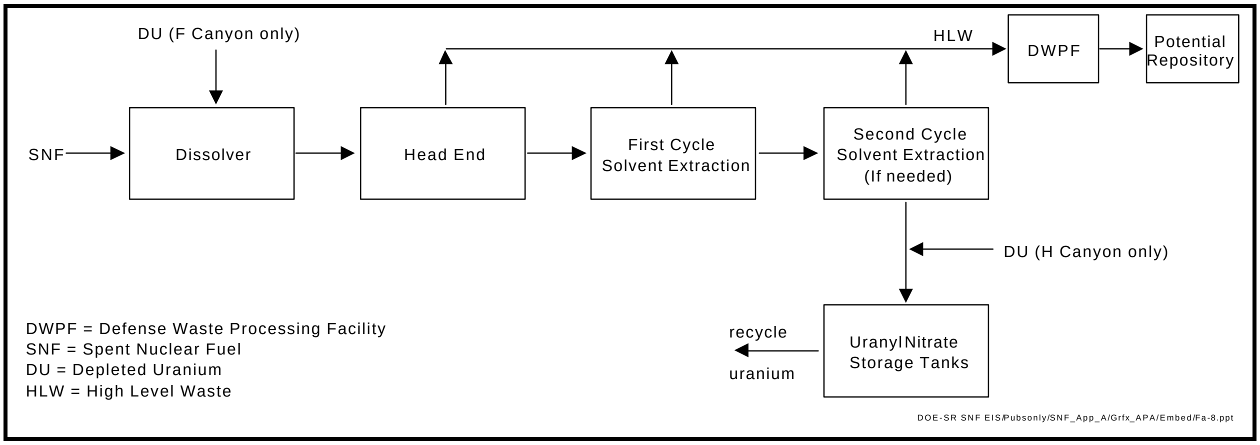

DWPF = Defense Waste Processing Facility
SNF = Spent Nuclear Fuel
DU = Depleted Uranium
HLW = High Level Waste

Figure A-8. Conventional Processing flow diagram.

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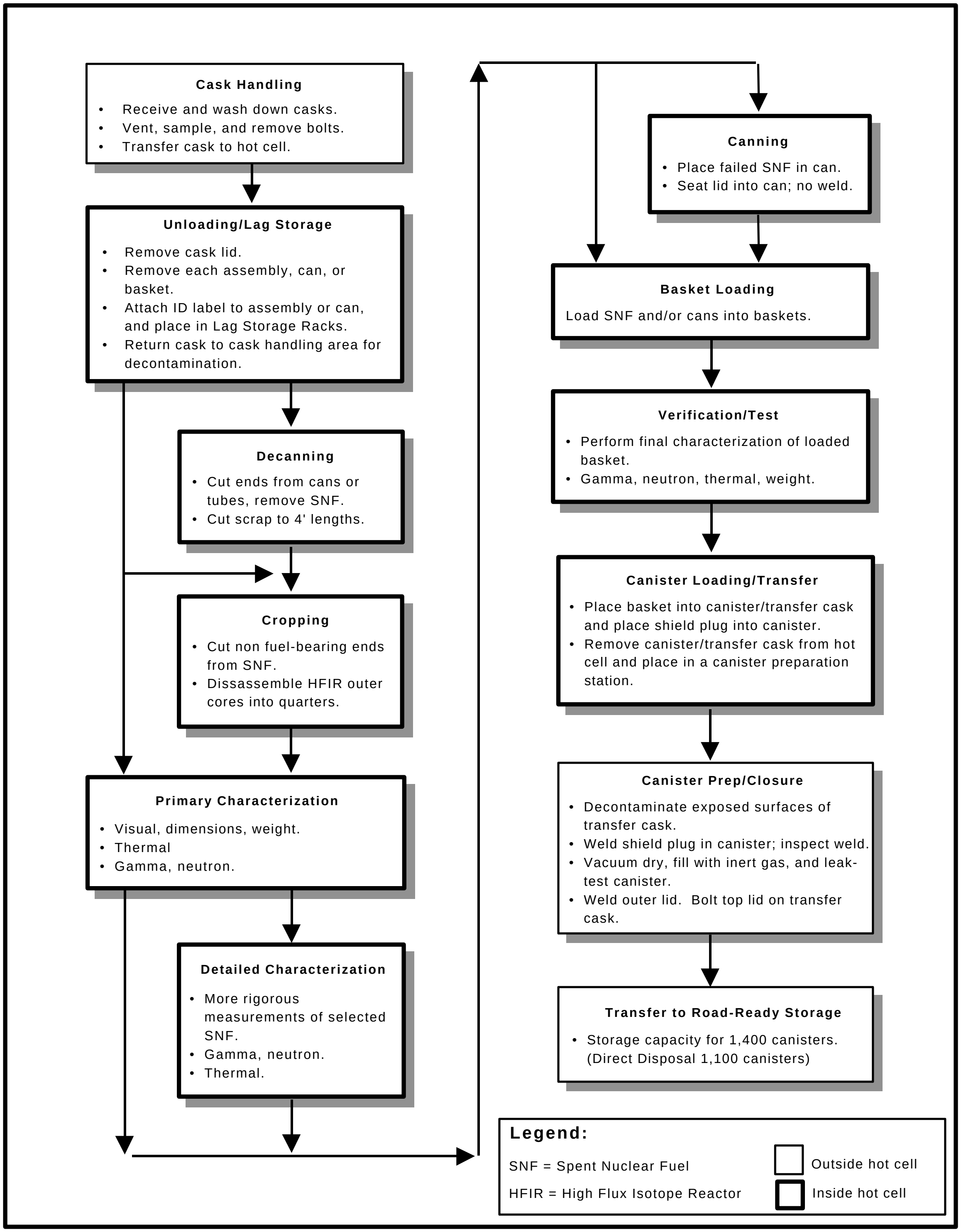

Cask Handling
- Receive and wash down casks.
- Vent, sample, and remove bolts.
- Transfer cask to hot cell.

Unloading/Lag Storage
- Remove cask lid.
- Remove each assembly, can, or basket.
- Attach ID label to assembly or can, and place in Lag Storage Racks.
- Return cask to cask handling area for decontamination.

Decanning
- Cut ends from cans or tubes, remove SNF.
- Cut scrap to 4' lengths.

Cropping
- Cut non fuel-bearing ends from SNF.
- Dissassemble HFIR outer cores into quarters.

Primary Characterization
- Visual, dimensions, weight.
- Thermal
- Gamma, neutron.

Detailed Characterization
- More rigorous measurements of selected SNF.
- Gamma, neutron.
- Thermal.

Canning
- Place failed SNF in can.
- Seat lid into can; no weld.

Basket Loading

Load SNF and/or cans into baskets.

Verification/Test
- Perform final characterization of loaded basket.
- Gamma, neutron, thermal, weight.

Canister Loading/Transfer
- Place basket into canister/transfer cask and place shield plug into canister.
- Remove canister/transfer cask from hot cell and place in a canister preparation station.

Canister Prep/Closure
- Decontaminate exposed surfaces of transfer cask.
- Weld shield plug in canister; inspect weld.
- Vacuum dry, fill with inert gas, and leak-test canister.
- Weld outer lid. Bolt top lid on transfer cask.

Transfer to Road-Ready Storage
- Storage capacity for 1,400 canisters. (Direct Disposal 1,100 canisters)

Legend:

SNF = Spent Nuclear Fuel

HFIR = High Flux Isotope Reactor

☐ Outside hot cell

▣ Inside hot cell

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Figure A-1. Direct Disposal/Direct Co-Disposal process flow diagram.

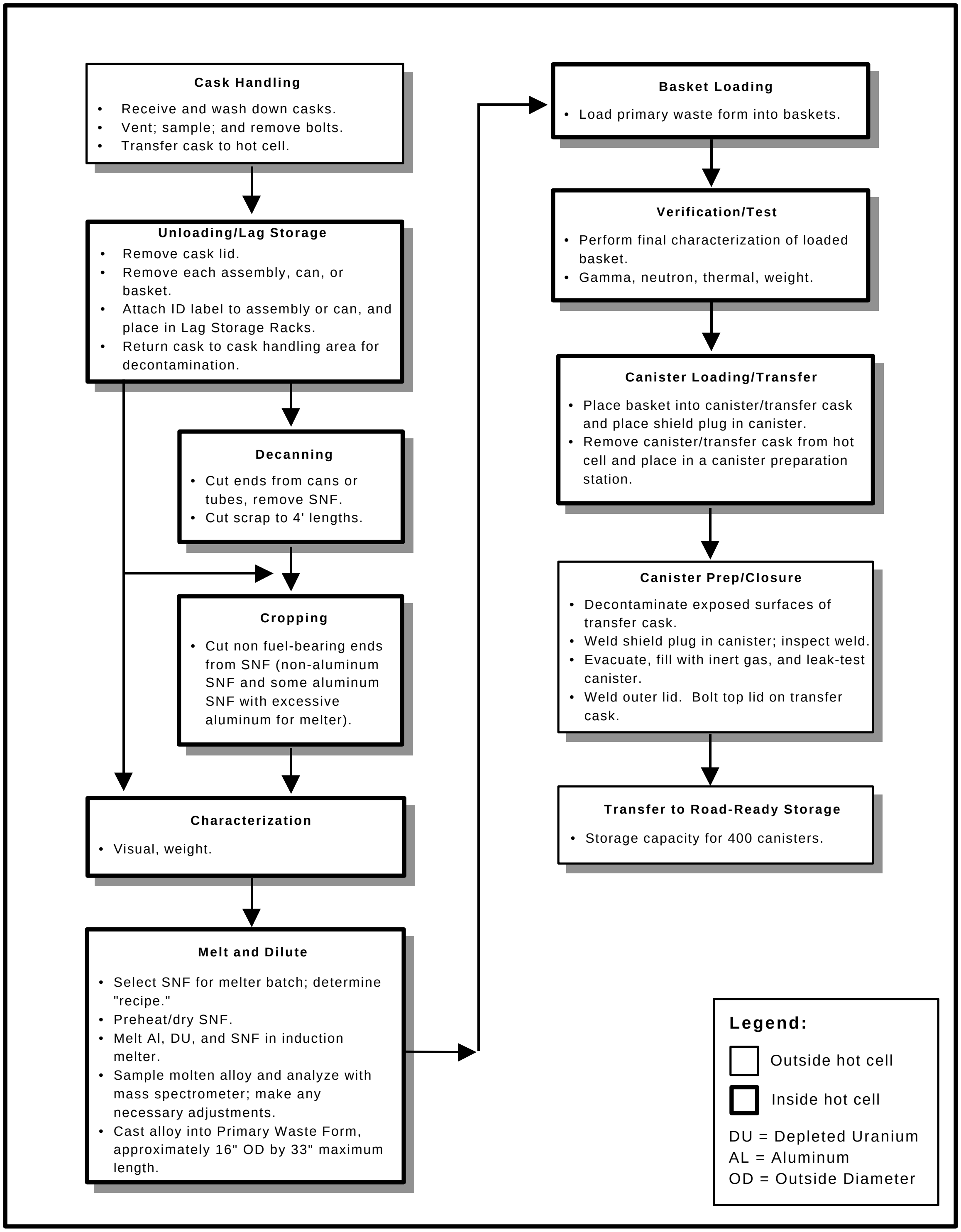

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Figure A-2. Melt and Dilute process flow diagram.

canister loading operation would be controlled to limit the fissile material in the canisters, in accordance with nuclear criticality considerations. The number of canisters produced for dry storage would be about 630 to be loaded as one per co-disposal overpack for repository disposal.

The primary advantage of Press and Dilute technology is its simplicity. However, the variable sizes of Group C SNF might make the technology unsuitable for those fuels without special disassembly before compaction. Particulate fuels (Group D) would not be amenable to pressing. Figure A-3 shows the Press and Dilute process flow diagram.

A.2.3 CHOP AND DILUTE

In the Chop and Dilute treatment, the shipping cask would be received and unloaded, and the SNF would undergo a limited characterization as described in Section A.2.2 Press and Dilute. The fuel assemblies would be cropped to eliminate most of the nonfuel structural components and reduce storage space. The assemblies would be vacuum-dried to remove free water.

The dried assemblies would be fed into a shredder. Similarly shredded depleted uranium-aluminum alloy would be combined with the shredded fuel to produce a mix of reduced enrichment. The shredded fuel would be placed into 17-inch (43-centimeter) diameter canisters, which would be filled with inert gas and welded closed. The canister loading would be controlled to limit the amount of fissile material in the canisters in accordance with nuclear criticality requirements. The number of canisters produced for dry storage and repository disposal would be about the same as for the Press and Dilute process (630).

The material resulting from Chop and Dilute would not be homogeneous and would result in a considerable amount of free space in each canister. The free space would contribute to an increase in the number of canisters required and could increase vulnerability to a nuclear criticality. In addition, the material could be pyrophoric. Because of these difficulties with Chop and Dilute, DOE considers it to be the least attractive of the three dilution technologies (Melt and Dilute, Press and Dilute, and Chop and Dilute).

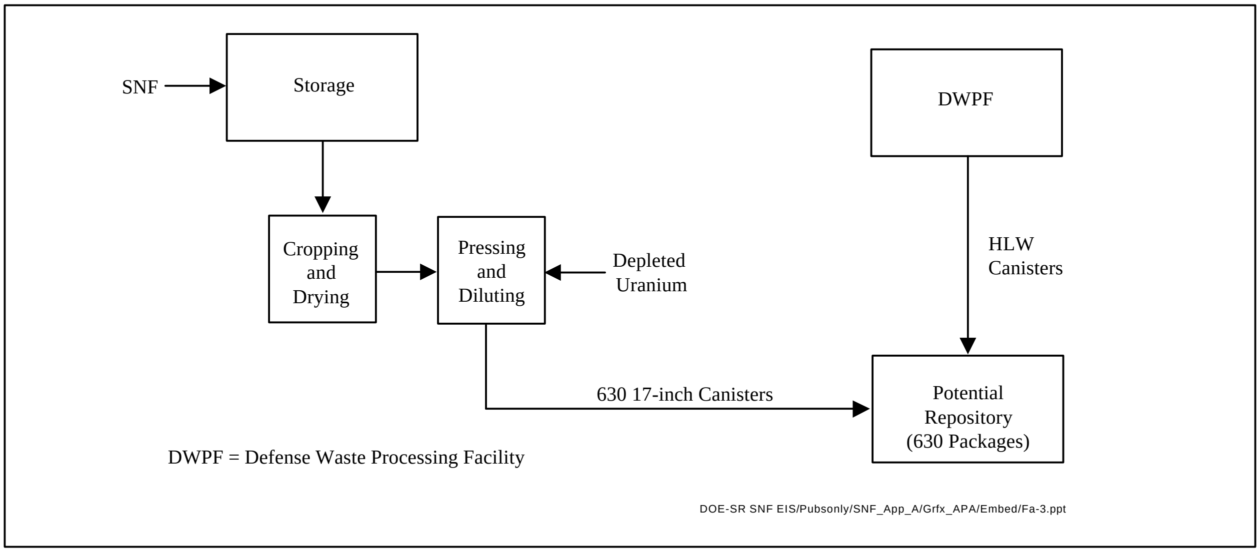

Figure A-3. Press and Dilute process flow diagram.

A.2.4 DISSOLVE AND VITRIFY

The SNF would be cropped and charged to an electrolytical dissolver similar to that used in H Canyon. The electrolyte solution would be nitric acid saturated with boric acid. The process would operate in a batch mode to ensure criticality control. Depleted uranium would be added, as needed, to reduce the uranium-235 enrichment to approximately 5 percent.

The dissolver solution would be transferred to a holding tank for chemical and radiological analyses to determine the need for any adjustments prior to the vitrification step. The solution then would be transferred to an electrically-heated melter, along with glass-forming chemicals. Several dissolver batches could be melted at once. The resulting molten glass, having been preanalyzed in the holding tank, should be of sufficient quality to be poured into canisters similar to those used at the Defense Waste Processing Facility. About 1,350 canisters would be produced for emplacement in about 270 repository waste packages.

Although DOE could perform dissolution using the existing equipment at H Canyon, the analysis in this EIS assumes the construction of a new Dissolve and Vitrify facility. Figure A-4 shows the Dissolve and Vitrify process flow diagram.

A.2.5 GLASS MATERIAL OXIDATION AND DISSOLUTION SYSTEM

The Glass Material Oxidation and Dissolution System (GMODS) converts SNF directly to borosilicate glass using a batch process. Criticality concerns are addressed by diluting the uranium-235 enrichment with depleted uranium and using boron oxide as a dissolving agent (boron is a neutron poison). Although the addition of depleted uranium and glass frit adds to the mass, the high-density, monolithic glass still would provide a smaller volume for dry storage than would Direct Co-Disposal.

The principal piece of equipment for GMODS would be an induction-heated cold-wall melter, which in commercial use converts corrosive or high-melting metals to ultrapure materials. The melter would be charged with a molten glass consisting of lead oxide and boron oxide. The lead oxide converts the metals in the SNF to oxides; oxides and amorphous materials tend to dissolve in molten glass, but metals do not. Boron oxide is a common agent for dissolving oxides into glass (e.g., welding slag). A problem

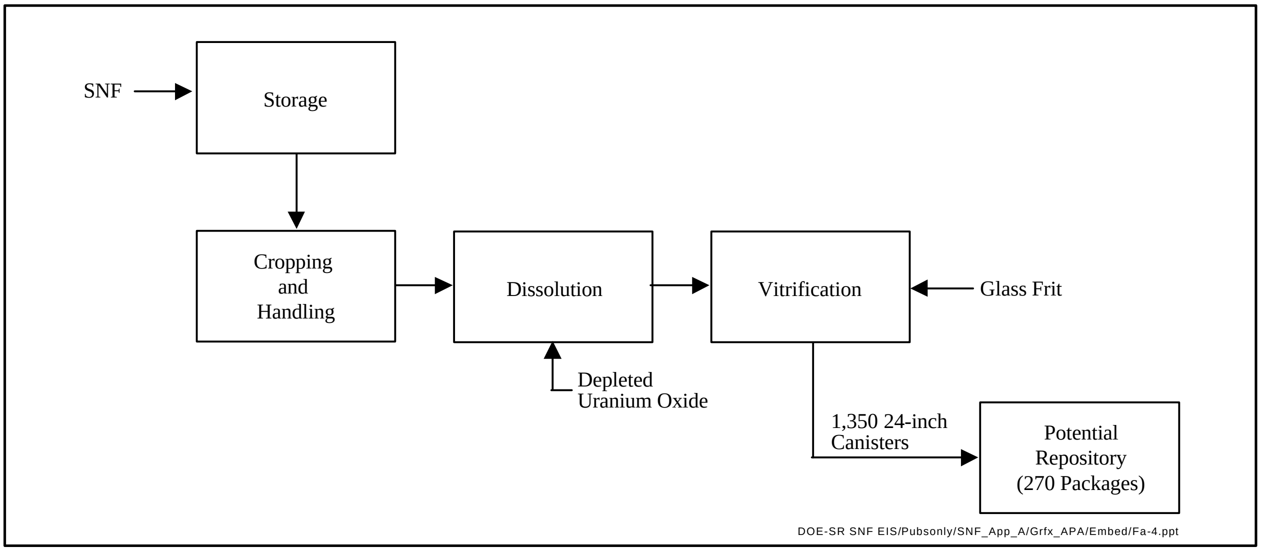

Figure A-4. Dissolve and Vitrify process flow diagram.

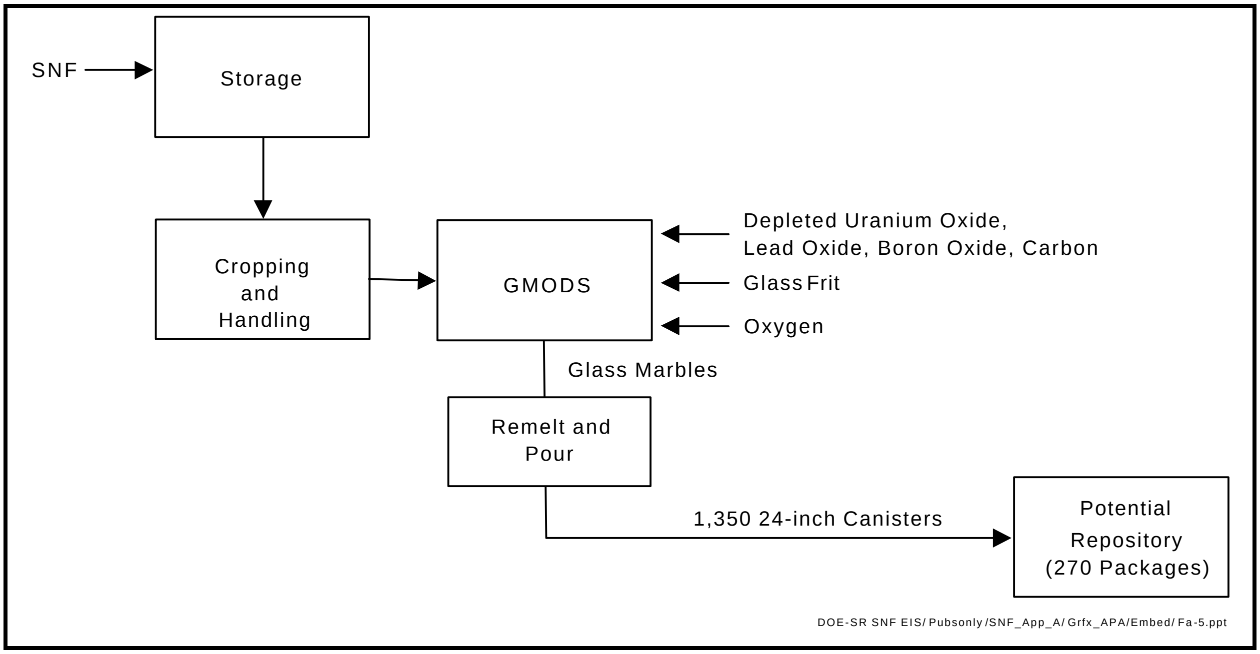

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Figure A-5. Glass Material Oxidation and Dissolution process flow diagram.

1,600°C (2,900°F) to melt and oxidize it in a rotating furnace. Ceramic material would be added as necessary with the mixture homogenized by the torch. When melting and oxidation were complete, the rotating furnace would slow and the melt would fall into molds prepared to receive it.

Some types of SNF might not require the addition of ceramic material to the process because the oxidation would produce a robust ceramic form from the fuel itself. Many metallic fuels would, however, need some ceramic addition. Depleted uranium could be added to the process in almost any form to reduce the uranium-235 enrichment. Criticality issues would be addressed by limiting the process to batch runs of preselected quantities of fissile material, by the addition of the depleted uranium, and by the addition of neutron poisons if necessary. The Plasma Arc treatment would produce about 490 canisters to be contained in 98 repository packages.

As with all processes that dissolve or melt the SNF, the Plasma Arc Treatment would produce radioactive offgases. These gases would be filtered and treated by appropriate means, with the filter and treatment media recycled into the plasma arc furnace for incorporation into the ceramic product. Figure A-6 shows the Plasma Arc Treatment process flow diagram.

A.2.7 ELECTROMETALLURGICAL TREATMENT

The Electrometallurgical Treatment process would adapt a technology under development at the Argonne National Laboratory for processing Experimental Breeder Reactor-II fuel and blanket assemblies. The process has been demonstrated for the stainless steel-clad uranium alloy fuels used in this reactor. The electrorefining process employs a technology used in industry to produce pure metals from impure metal feedstock. The feasibility of the Electrometallurgical Treatment for aluminum-based fuels has been tested in the laboratory and is theoretically possible as conducted in the following two stages. An electrorefiner facility is available at the Idaho National Engineering and Environmental Laboratory to test development concepts.

Preparation

Before electrorefining, the fuel would be cropped and the end fittings discarded. The fuel assemblies would be compacted and melted with

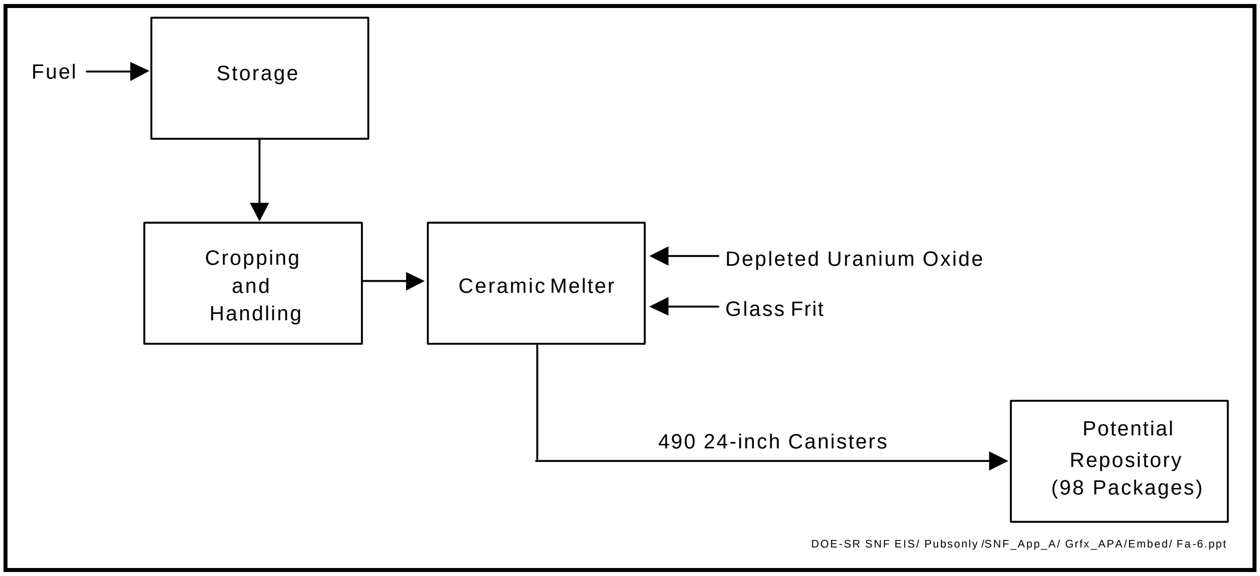

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Figure A-6. Plasma Arc Treatment process flow diagram.

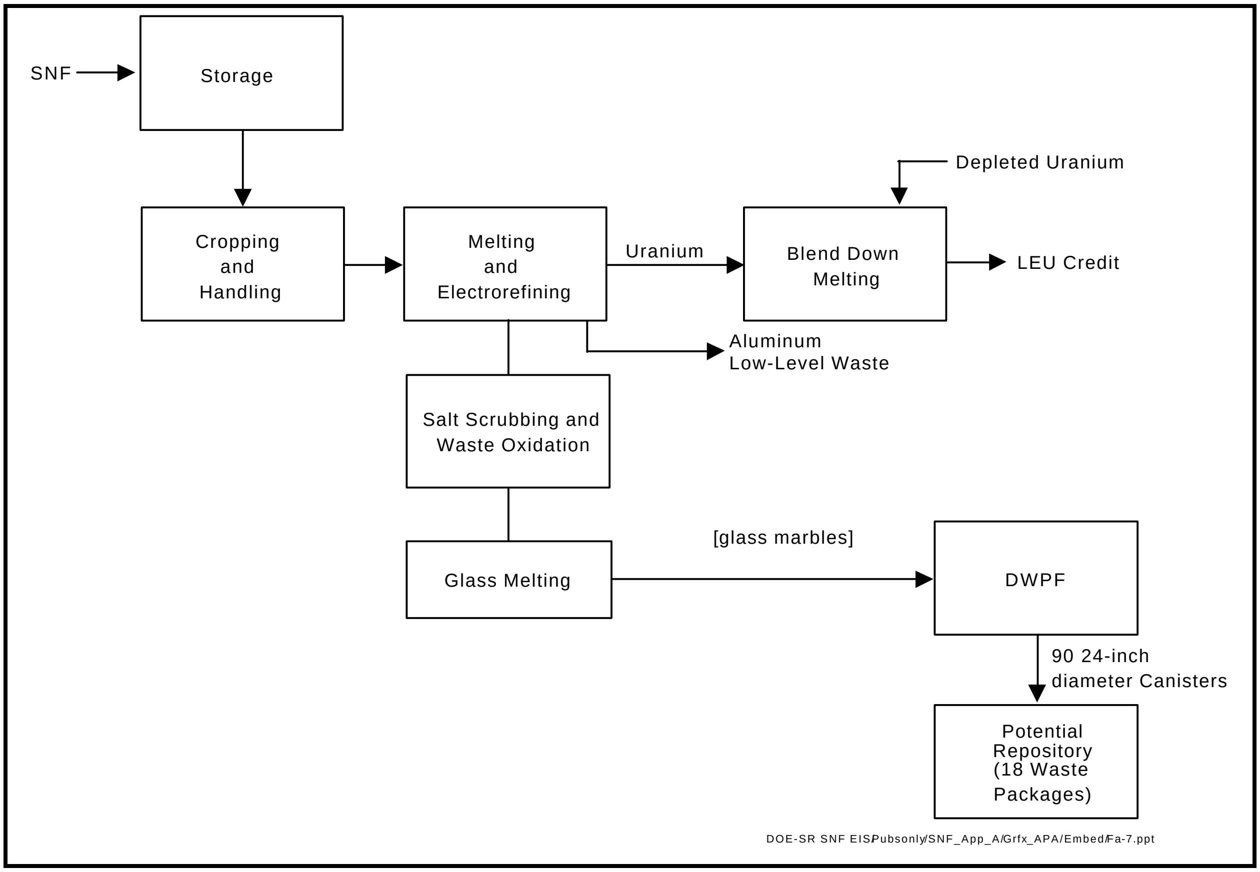

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Figure A-7. Electrometallurgical Treatment process flow diagram.

A.2.8 TECHNOLOGIES NOT ANALYZED

This section describes technologies that DOE considered but did not analyze further in the EIS because the technologies need further research to demonstrate they are technically viable and cost effective. These technologies have not undergone bench-scale demonstrations.

Chloride Volatility

Chloride volatility is an advanced treatment technology being investigated at the Idaho National Engineering and Environmental Laboratory. The process segregates major nonradiological constituents from SNF for the purpose of volume reduction, and isolates the fissile material to produce a glass or ceramic waste form.

The process is based on completely volatilizing the fuel elements and separating the gaseous constituents. The fuel would react with chlorine gas at a temperature greater than 1,200°C (2,200°F) to produce volatile chlorides. The fission products and transuranics would be separated by passing the gas through molten zinc chloride in a counter-current scrubber. The gases minus the fission products and transuranics would flow through a series of condensers to remove chloride compounds by fractional distillation. The series of uranium chlorides could be recovered separately, if desired.

The molten zinc chloride would be regenerated by vacuum distillation to recover it for recycle. The fission product and transuranic residue would be converted to oxides or fluorides by fluorination for vitrification and melting with glass frit additives. As an alternative, the residues could be oxidized by boric acid at high temperatures. The transuranics could be separated from the fission products by solvent extraction if separate disposal were necessary.

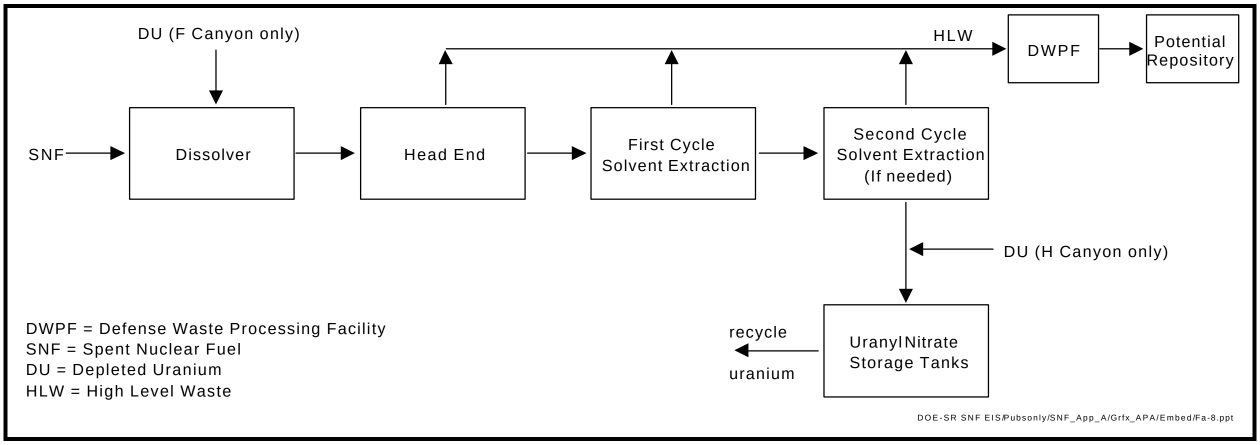

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Figure A-8. Conventional Processing flow diagram.

APPENDIX B

IDENTIFICATION AND RESOLUTION OF SAVANNAH RIVER SITE SPENT NUCLEAR FUEL VULNERABILITIES

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APPENDIX B. IDENTIFICATION AND RESOLUTION OF SAVANNAH RIVER SITE SPENT NUCLEAR FUEL VULNERABILITIES

B.1 Purpose

The end of the Cold War brought an end to active efforts in the United States to produce nuclear weapons materials such as plutonium. As a consequence, nuclear materials produced for weapons have been stored temporarily for prolonged periods in systems and under conditions not originally designed for long-term storage. Prolonged storage in systems and under conditions designed for short-term storage has degraded the integrity of some of these materials and has led to concerns about safety. These concerns have been documented in reports by both the U.S. Department of Energy (DOE) and the Defense Nuclear Facilities Safety Board (DNFSB). The purpose of this appendix is to provide a compilation of spent nuclear fuel (SNF) storage problems (vulnerabilities) specific to the Savannah River Site (SRS), their recommended corrective actions, and the current status of those corrective actions.

B.2 Introduction and Background

For about 30 years, DOE operated heavy-water reactors at SRS for the production of defense nuclear materials. Low-temperature reactor operation allowed the use of aluminum-clad, aluminum-alloy fuel and aluminum-clad target materials. This reactor design facilitated both fuel and target fabrication and subsequent processing. At the end of a reactor cycle, the fuel and targets were normally discharged to cooling basins and stored for as long as 18 months prior to processing.

In the past, the SRS processed SNF and other reactor-irradiated nuclear materials (RINM) to recover plutonium, tritium, and other isotopes. In April 1992, with chemical separations activities already temporarily suspended, DOE implemented a decision to phase out defense-related chemical separation activities at the SRS. Processing of the "in-process" RINM was not completed. Facilities designed, constructed, and operated to store RINM for relatively short periods had to store it for relatively long periods pending decisions on the disposition of the materials.

B.3 Spent Fuel Working Group

In August 1993, the Secretary of Energy commissioned a comprehensive baseline assessment of the environmental, safety, and health (ES&H) vulnerabilities associated with the storage of reactor-irradiated nuclear materials in the DOE complex. In October 1993, a multidisciplinary Spent Fuel Working Group, comprised of DOE and contractor employees, assessed 66 facilities on 11 sites. Eight SRS facilities that contained RINM were assessed. The facilities included both wet and dry storage systems. The assessment's objective was to provide an itemized inventory of RINM and an initial assessment of ES&H vulnerabilities associated with the current storage and handling of these materials.

DOE defined vulnerabilities as conditions or weaknesses that could lead to radiation exposure to the public, unnecessary or increased exposure to the workers, or release of radioactive materials to the environment. The loss of institutional controls, such as cessation of facility funding or reductions in facility maintenance and control, could also cause vulnerabilities. Reactor-irradiated nuclear material was defined as spent nuclear fuel and irradiated nuclear targets from production and research reactors; however, it did not include fuel currently in active reactors or irradiated structural materials (other than fuel cladding).

The assessment focused on determining ES&H vulnerabilities and presenting factual information. In general, DOE did not identify or recommend future corrective actions, but did assess corrective actions already under way. Evaluations were made of facilities, structures, systems, operating conditions, and procedures necessary to protect the workers, the public, and the environment during the storage and in-facility handling of reactor-irradiated nuclear material.

On December 7, 1993, the Working Group released *Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and Other Reactor Irradiated Nuclear Materials and Their Environmental, Safety, and Health Vulnerabilities* (DOE 1993) ("The Working Group Report," Volumes I, II and III). Volume I summarized the findings, including: (1) the characteristics and inventory of reactor-irradiated nuclear material; (2) ES&H vulnerabilities associated with different storage options; (3) five generic issues common to many storage facilities; and (4) identification of eight facilities requiring priority management attention, including the SRS L- and K-Reactor Disassembly Basins.

Volume II of the Working Group Report contains Working Group Assessment Team reports for each site, Vulnerability Development forms, and documents used by the Working Group Assessment Team as information sources. Volume II categorized vulnerabilities based on the period during which it was recommended that the vulnerability be addressed. For each of the eight SRS facilities, vulnerabilities were grouped into one of three time periods for management attention: less than 1 year, 1 to 5 years, and more than 5 years.

Volume II identified 21 SRS vulnerabilities. A twenty-second vulnerability was identified later. When DOE reviewed the ES&H vulnerabilities, it determined that two (SRS-2 and SRS-3) were not vulnerabilities and obtained agreement from the working group assessors. Table B-1 lists the SRS vulnerabilities and their assigned priorities.

Fifteen vulnerabilities warranting priority management attention, including one potential vulnerability, were identified for the SRS L-, K-, and P-Reactor Disassembly Basins. Four major vulnerabilities and one generic vulnerability were identified for the Receiving Basin for Offsite Fuel (RBOF). The *Reactor Division Disassembly Basin Management Plan* (Burke 1993) addressed and provided resolution of the vulnerabilities identified for the reactor disassembly basins and RBOF. A February 21, 1997 memorandum reports on the corrective action closure package for the reactor disassembly basins and RBOF vulnerabilities (Burke 1997).

In February 1994, DOE released the first phase of a three-phased plan to remedy vulnerabilities associated with the storage of spent fuel and irradiated materials. The *Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities, Phase I* (DOE 1994a) described actions that had been completed or for which no major funding or policy issues existed. After the Phase I report was issued, DOE resolved most funding issues associated with SNF vulnerabilities. The Phase II Plan of Action (DOE 1994b), published in April 1994, was the product of follow-on work to the Phase I report.

The Phase III Plan of Action (DOE 1994c), the second update to the original Plan of Action, was issued in October 1994. The Phase III report focused on the resolution of critical policy issues and incorporated stakeholder comments on the original Plan of Action and the first update. Table B-2 lists the Phase I, II, and III corrective action plans and their reported status.

B.4 Defense Nuclear Facilities Safety Board Recommendation 94-1

In May 1994, the DNFSB issued Recommendation 94-1, *Improved Schedule for Remediation in the Defense Nuclear Facilities Complex* (DNFSB 1994). The Board expressed its concern that imminent hazards could arise during the next 2 to 3 years unless problems related to the state of reactor-irradiated nuclear material remaining from the production of nuclear weapons were resolved.

Table B-1. SRS vulnerabilities identified in Spent Nuclear Fuel Working Group Report.

	Priority[a]		
Vulnerability	1	2	3
SRS-01, L-Reactor Disassembly Basin: Potential unmonitored build-up of radionuclide and/or fissile materials in sand filters.	√		
SRS-04, L-Reactor Disassembly Basin: Lack of authorization basis in operating sand filter cleanup system for L-Area Disassembly Basin.	√		
SRS-05, L-Reactor Disassembly Basins: Corrosion of aluminum- clad fuel, targets, and components.			√
SRS-06, L-Reactor Disassembly Basin: Cesium-137 activity level in L-Basin.	√		
SRS-07, L-Reactor Disassembly Basin: Determine if gas bubble release above the bucket storage area is a potential hazard at L-Reactor.	√		
SRS-08, K-, L-, and P-Reactors: Lack of reactor authorization basis.	√		
SRS-09, L-Reactor Disassembly Basin: Corrosion of Mark-31A and B target slugs in K- and L-Disassembly Basins.	√		
SRS-10, P-Reactor Disassembly Basin: Hoist rod corrosion.		√	
SRS-11, K- and L-Reactor Disassembly Basin: Reactor disassembly basin safety analysis envelope.	√		
SRS-12, L-Reactor Disassembly Basin: Inadvertent flooding of L-Reactor Disassembly Basin.	√		
SRS-13, K-Reactor Disassembly Basin: Inadvertent flooding of K-Reactor Disassembly Basin.	√		
SRS-14, P-Reactor Disassembly Basin: Inadvertent flooding of P-Reactor Disassembly Basin.	√		
SRS-15, RBOF, P-, L-, C-, and R-Reactors: Conduct of Operations at reactor facilities and RBOF. (NOTE: RBOF is a less-than-1-year vulnerability.)	√		
SRS-16, RBOF: Inadequate tornado protection at RBOF.		√	
SRS-17, RBOF: Seismic vulnerability of RBOF.		√	
SRS-18, H-Canyon: Seismic vulnerability of H-Canyon.			√
SRS-19, F-Canyon: Seismic vulnerability of F-Canyon.			√
SRS-20, K-, L-, and P-Reactor Disassembly Basins and RBOF: Inadequate leak detection system in the underground water-filled RINM storage basin.		√	
SRS-21, K-, L-, and P-Reactor Disassembly Basins: Inadequate seismic evaluation and potential inadequacies of structures, systems and components to withstand a Design-Basis Earthquake.	√		
SRS-22, R-Area: Potential buried spent nuclear fuel.		√	

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a. **Priority 1:** Vulnerabilities identified by the Working Group as warranting immediate management attention.
Priority 2: Vulnerabilities requiring action within 1 year.
Priority 3: Vulnerabilities requiring action within 1 to 5 years (DOE 1994a).

Table B-2. Status of Savannah River Site Vulnerability Corrective Action Plans.

Identified vulnerabilities	Corrective action plans	Status
SRS-1: Potential unmonitored buildup of radionuclide/fissile material in sand filters. (L-Basin)	**SRS-1a:** Perform characterization analysis of isotopes in existing sand filter system.	**Completed 5/95.** Characterization analysis completed.
	SRS-1b: As part of the Basis for Interim Operations development, perform safety analysis for buildup of fissile material in sand filter system and potential for criticality in filters (see items 8 and 11).	**Completed 7/96.** The criticality safety evaluation determined that there is not an identifiable mechanism by which a critical configuration could be assembled in the disassembly basin sand filter. The L-Basin Basis for Interim Operations (WSRC 1996) concluded that a criticality is not a credible event in the sand filter.
SRS-4: Lack of characterization and updated safety analysis for fissile material in sludge on basin floors and sand filter cleanup systems (K- and P-Basins). (generic issue)	**SRS-4a:** Complete development and application of technologies required (at L-Basin) for characterization and analysis, removal, and disposal of sludge from L-Basin.	
	SRS-4a.1: Complete characterization of sludge from L-Basin.	**Completed 9/93.** Sludge analysis completed.
	SRS-4a.2: Complete characterization of sludge from K-Basin.	**Completed 6/93.** Sludge analysis completed.
	SRS-4a.3: Complete characterization of sludge from P-Basin.	**Completed 5/93.** Sludge analysis completed.
	SRS-4a.4: Complete removal of sludge from L-Basin.	**Phase I (Sludge Consolidation):** Completed 3/95.
	NOTE: The ability to maintain excellent basin water quality in the presence of sludge has been demonstrated, eliminating the urgency to consolidate and remove the sludge to prevent further corrosion of stored fuel.	**Phase II (Sludge Removal):** Completed 1999.
	SRS-4a.5: Complete removal of sludge from K-Basin.	**Phase I (Sludge Consolidation):** Completed 3/98 (Smith 1998).
	NOTE: The ability to maintain excellent basin water quality in the presence of sludge has been demonstrated, eliminating the urgency to consolidate and remove the sludge to prevent further corrosion of stored fuel.	**Phase II (Sludge Removal):** Cancelled 4/15/98 (Conway 1998).
	SRS-4a.6: Complete removal of sludge from P-Basin.	**Cancelled 4/15/98** (Conway 1998).
	SRS-4b: For characterization and safety analysis of fissile material in sand filters (see item 1).	**Completed 7/96.** See SRS 1a and 1b.
SRS-6: Cesium-137 activity level in L-Basin water is approaching administrative limits.	**SRS-6:** Utilize Zeolite in portable ion exchange system to lower Cesium-137 levels in L-, K-, and P-Basins.	**Completed 7/96.** Zeolite was used in a portable ion exchange system to lower the Cesium-137 levels in L- and K-Basins. P-Basin zeolite cancelled due to fuel consolidation.

TC

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
SRS-5, 7, and 9: Aluminum-clad fuel and targets are severely corroded, releasing fission products and fissile material to the pool water. Gas bubble release above the bucket storage area at L-Basin might be a potential hazard. (generic issue).	**SRS 5,7,9a.01:** Modify fuel hangers to provide redundancy against fuel falling to basin floor.	**Completed 1/93.** No fuel on fuel hangers in P-Basin; fuel in L-Basin Vertical Tube Storage has been removed from hangers and placed in horizontal tube storage; fuel hangers in K-Basin Vertical Tube Storage were inspected and found to be in good condition; the fuel was relocated to horizontal tube storage in July 1997.
	SRS-5,7,9a.02: Develop and implement corrosion surveillance program.	**Completed 1/93.** The corrosion surveillance program is summarized in *Corrosion Surveillance in Spent Fuel Storage Pools* (Howell 1996).
	SRS-5,7,9a.03: Complete criticality studies in progress to support the transfer of reactor components from vertical to horizontal storage.	**Completed 1/94.** Criticality studies to support the transfer of reactor components are documented in *100 Area Irradiated Fuel Consolidation and Horizontal Storage Criticality Concerns* (Reed 1994).
	SRS-5,7,9a.04: Design and construct racks for horizontal three-deep storage.	**Completed 6/95.** Horizontal storage rack fabrication and installation completed under Project S-5982 (Guy 1995).
	SRS-5,7,9a.05: Reorient fuel currently stored vertically to the three-deep horizontal array configuration at L-Basin.	**Completed 12/95.** *Storage Solution for Fuel Tubes in the L-Area Vertical Tube Storage* (Guy 1993) provided the engineering direction for the reorientation activities. Completion of reorientation is documented in *L-3.3 Fuel Bundling* (Holmes 1995), and *Disassembled Component Log - Fuel Bundling Station* (WSRC 1995a).
	SRS-5,7 9a.06: Reorient fuel currently stored vertically to the three-deep horizontal array configuration at K-Basin.	**Completed 7/97** (Smith 1998).
	SRS-5,7,9a.07: Modify water chemistry of cleaned basins through the intensive use of portable deionizers (vendor supplied, shock deionization) at L-Basin.	**Completed 9/95.** Deionization reduced conductivity to 10 µs/cm.
	SRS-5,7,9a.08: Modify water chemistry of cleaned basins through the intensive use of portable deionizers (vendor supplied, shock deionization) at K-Basin.	**Completed 1/96.** Deionization reduced conductivity to 10 µs/cm.
	SRS-5,7,9a.09: Modify water chemistry of cleaned basins through the intensive use of portable deionizers (vendor supplied, shock deionization) at P-Basin.	**Canceled 7/95.** Deionization canceled due to P-Basin fuel consolidation (DOE 1995a).

	SRS-5,7,9a.10: Provide deionized makeup water systems for the basins.	**Completed 10/95.** Systems installed under Project S-5839. Functional performance requirements are documented in *Disassembly Basins Upgrades* (WSRC 1995b).

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
	SRS-5,7,9a.11: Assess the hazard of gas releases as a result of the corroding material in the bucket storage area at L-Basin.	**Completed 8/95.** Evaluation of gas releases study indicated the exposure potential from leaking targets in the basin is insignificant (Hochel 1995).
	SRS-5,7,9a.12: Maintain basin water chemistry through the application of additional dedicated and upgraded deionizers and regeneration capabilities.	**Completed 6/96.** Continuous deionization systems were installed and tested for K- and L-Basins under Project S-5839 (New 1996a).
	SRS-5,7,9a.13: Assess deionizer regeneration at RBOF facilities to support timely regeneration of L-, K-, and P-Basin ion exchange resins.	**Completed 10/94.** Assessment of deionizer regeneration facilities at RBOF was documented in *Division Critical Item - RBOF Regeneration System Improvements* (Cederdhal and Freeman 1994).
	SRS-5,7,9a.14: Complete modifications to regeneration equipment at RBOF if determined appropriate by assessment.	**Completed 6/98** (Smith 1998).
	SRS-5,7,9a.15: Complete placement of MK-31 slugs stored in L-Basin into containment boxes to minimize the spread of fission and corrosion products.	**Completed 1/94.** Containment program was developed and implemented to reduce the spread of contamination from the corroding target slugs. Subsequently, all targets were removed from L-Basin and processed.
	SRS-5,7,9b.1: Develop acceptance criteria and validated heat transfer models for highly enriched uranium aluminum-clad fuel.	**Completed 3/96.** Acceptance criteria established and documented (Sindelar et al. 1996).
	SRS-5,7,9b.2: Complete development of generic dry storage procurement specification.	**Completed 12/95.** Specifications completed and documented (New 1995).
	SRS-5,7,9b.3: Complete preconceptual design studies for dry storage option.	**Completed 3/12/98** (G-CDR-L-00001) (Smith 1998).
	SRS-5,7,9b.4: Complete Environmental Impact Statement (EIS) for dry storage.	**Ongoing, scheduled for 04/00.** Notice of Intent issued by DOE to prepare an EIS on management of aluminum-clad fuel at SRS (61 FR 69085). The scope of the EIS will include an assessment of the impacts associated with construction and operation of a dry storage facility. TC
	SRS-5,7,9b.5: Complete civil structural design for dry storage.	**Preliminary design work is scheduled for FY03, followed by final design in FY04 and FY05.** TC

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
SRS-8 and 11: Lack of adequate authorization bases (and operating procedures) including updated and approved SAR that addresses long-term storage of RINM and accident mitigation.	**SRS 8,11:** Complete Basis for Interim Operation (BIO) per DOE 5480.23 and Facility Hazards Assessment per DOE 5500.3A, currently under development for K-Basin (anticipated to be bounding for L- and P-Basins). BIO development will include: - Performance of safety analysis for buildup of fissile material in sand filter system and evaluation of potential for criticality in filters. - Evaluation of basin ventilation requirements. - Evaluation of potential cask drop hazards. - Evaluation of potential hazards associated with basin flooding. - Evaluation of capability to maintain cooling water during seismic events and other credible initiating events. Issues include a loss of emergency cooling source and results of percolation studies.	**Completed 9/95.** Completed the BIO per DOE Order 5480.23 for K-Basin, bounding for L- and P-Basins, and forwarded to DOE for review in 8/94; DOE approved the BIO in 3/95 (WSRC 1995c). Completed the Facility Hazards Assessment per DOE Order 5500.3A for K-Basin, bounding for L- and P-Basins (WSRC 1995d).
SRS-10: Failure of severely corroded hoist rod bolts on the twin hoist in the P-Basin could result in a fuel drop and subsequent fission product release or violation of criticality safety spacing requirements.	**SRS-10:** Conduct hoist assembly load test based on revised preventative maintenance procedure. This does not pose a criticality concern. Administrative controls and geometric constraints dismiss the risk of criticality. The hoist assembly load test will be conducted prior to use of the hoist. No use of the hoist is contemplated for the foreseeable future and it is not cost-effective to perform the load test until use of the hoist is required.	**Completed 11/95.** The Twin Hook Hoist was replaced in P-Area and the replacement hoist load tested using Work Request No. BHBRS (WSRC 1995e).
SRS-12-14: Flooding of basins initiated by human error or seismic event (affecting the makeup system) could result in basin overflow with resultant fission product release to environment.	**SRS-12-14:** Evaluate potential hazards associated with basin flooding as part of the Basis for Interim Operation development. (See Item SRS 8,11).	**Completed 3/95.** The potential hazards associated with basin flooding were evaluated as part of the BIO development, which concluded that 105-K, -L and -P buildings are not subject to flooding (WSRC 1995c).
SRS-15: Conduct of operations emphasis on the extended role of SRS fuel storage basins is necessary. (Vulnerability applicable to RBOF and P-, K-, L-, C-, and R-Reactors).	**SRS-15:** Conduct training to emphasize extended storage of production reactor fuels and target materials.	**Completed 2/94.** Training specifically designed to address the concerns with extended storage of SNF was conducted. Individual training records are maintained in Building 704-24K.

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
SRS-16: The roof over the cask basins and the transite walls of the RBOF facility provide inadequate tornado missile protection.	**SRS-16:** Complete a detailed structural assessment for design-basis hazards (seismic, tornado, etc.). See Item SRS-17 actions. A detailed structural assessment for the design-basis hazards for the facility will be part of the safety analysis report upgrade. RBOF Technical Safety Requirements (TSRs) to be submitted to DOE in FY95. The new Safety Analysis Report will be complete in FY96.	**Completed.**. A detailed structural assessment for the design basis hazards for the facility (seismic, tornado, etc.) has been completed as part of the safety analysis upgrade. The analysis includes the basins, the above-grade walls, roof, and the storage racks. Following the analysis, limited modifications were performed. The analysis has shown RBOF meets DOE requirements for natural phenomenon hazards. (New 1996b). The Safety Analysis Report and TSRs have been approved by DOE.
SRS-17: Since the initial design, there has been no deterministic seismic evaluation of the facility. A seismic event could damage masonry walls above the pool, the sole significant makeup water line penetrating the facility foundation, and unanchored chemical storage tanks/piping adjacent to the facility, adversely affecting the RBOF. (Storage racks, although anchored to the floor and wall of the basin, are not seismically qualified.)	**SRS-17:** Complete a detailed assessment in conjunction with the efforts to upgrade the Safety Analysis Report. A detailed structural assessment for the design-basis hazards for the facility (seismic, tornado, etc.) will be part of the safety analysis report upgrade. The analysis will include the basins, the above-grade walls, roof and storage racks. In the interim, the existing safety analysis bounds radiological releases from all credible scenarios because it is extremely conservative. The new Safety Analysis Report will be complete in FY96.	**Completed.** Actions accomplished in response to SRS-16 have also completed the corrective actions required for SRS-17. (See Issue SRS-16.) The Safety Analysis Report was developed and submitted for DOE-SR review and approval in August 1997 (Smith 1998). Safety Analysis Report and TSRs have been approved by DOE.
SRS-18: Potential seismic vulnerability of H-Canyon because the facility does not meet current seismic design standards. Initial seismic calculations indicate that portions of the H-Canyon facility that house the target storage vault are not structurally adequate, which could result in a direct release path.	**SRS-18a:** Complete detailed seismic structural assessment along with Safety Analysis Report upgrade. **SRS-18b:** Complete development of new Technical Safety Requirements.	**Completed.** Analysis has shown H-Canyon meets DOE requirements of seismic resistance (Alm 1997). Also, a recent review of U.S. Geological Survey hazard maps indicates the map results would have no significant impacts on hazard results used for H-Canyon seismic analysis work (WSRC 1997a). **Completed.** Technical Safety Requirements were approved 4/97.
SRS-19: Potential seismic vulnerability of F-Canyon because the facility does not meet current seismic design standards. Initial seismic calculations indicate that portions of the F-Canyon facility that house the fuel storage vault are not structurally adequate, and that could result in a direct release path from the facility in the event of a Design Basis Earthquake. A criticality could result from seismically-induced damage to the storage racks, which could result in additional radiation release.	**SRS-19:** Complete detailed seismic analysis for F-Canyon.	**Completed 8/96.** The detailed seismic calculations were completed in 7/96. DOE-SR sent recommendations to DOE Headquarters to resume operations because results of seismic calculations were favorable. DOE Headquarters approved resumption of F-Canyon operations, which started on 8/26/96 (Alm 1996a). Also, a recent review of U.S. Geological Survey hazard maps indicates the map results would have no significant impacts on hazard results for F-Canyon seismic analysis work (WSRC 1997a).

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
SRS-20: Lack of adequate leak detection system for the storage basins. The current leak detection method is not sufficiently sensitive to detect small leaks.	**SRS-20a (for L-, K-, and P-Basins):** Install additional monitoring wells (two per basin) for L-, K-, and P-Basins to ensure detection of basin leakage.	**Completed 3/95.** Monitoring wells for L-, K-, and P-Basins completed 3/2/95 (Burbage 1995). Installation completed under Project S-5839 (New 1996a).
	SRS-20a (for RBOF): Perform studies of other beneficial methods of leak detection.	**Completed 1/95.** Overall H-Area groundwater effects are monitored and reported in accordance with applicable requirements (Clark 1994; Burbage 1995).
	SRS-20b (for L-, K-, and P-Basins): Improve the basin leak detection threshold by comparing trends in radionuclide concentration observed in the monitoring wells and the basins. Chemical constituents of samples from the monitoring wells and the basins will be monitored and trended for comparison purposes.	**Completed 12/96.** Basin level trending capabilities improved with the installation of upgraded simple level monitoring instruments and a makeup water system flow totalizer. Engineering initiated basin level trending, which provides more accurate monitoring of changes in the basin level. Monitoring wells downgradient of the basins have improved the dispersion/dilution models. Comparison of monitoring well chemical constituent trend data to basin water data has been initiated; monitoring wells are sampled monthly; evaluations of the radionuclide concentrations are issued in a quarterly report (Burbage 1996).
	SRS-20b (for RBOF): Evaluate and if necessary install improved level detection.	**Completed 1/95.** A study of the beneficial methods of leak detection included a review of the level monitoring capability used at the West Valley facility in New York. No significant benefits from implementation of this system were identified. Overall H-Area groundwater effects are being monitored and reported in accordance with applicable requirements. The RBOF level detection has been determined to be adequate (Clark 1994; Burbage 1995).
	SRS-20c (for L-, K-, and P-Basins): Evaluate the need for improved level detection system to provide more accurate monitoring of changes in basin level. (The accuracy of the current basin level indication is within 7,570 liters [2,000 gallons]).	**Completed 12/94.** Several options for level and leak detection systems were evaluated. DOE determined that the installation of an upgraded, simple level monitoring instrument coupled with the new makeup water and monitoring well system will provide an adequate cost-effective basin water inventory information system (New 1994).

Table B-2. (continued).

Identified vulnerabilities	Corrective action plans	Status
SRS-21: Inadequate seismic evaluation and potential inadequacies of structures, systems, and components to withstand a Design-Basis Event. The potential exists for: the failure of basin expansion joints and water stops, causing a release of radioactive materials to the environment; the failure of vertical tube storage frames or a load drop onto fuel assemblies causing damage or reconfiguration of fuel and possible criticality, due to a seismic event. (generic issue)	**SRS-21a:** Complete soil stability assessment for input to seismic analyses for L-, K-, and P-Basin that is in progress.	**Completed 10/94.** Soil stability assessments for K- and P-Basins are not required (Burke 1994). Geotechnical investigation into L-Areas is ongoing.
	SRS-21b: Complete seismic evaluations if determined to be necessary as a result of the Basis for Interim Operations development, or soil stability assessments for L-, K-, and P-Basin. A recent assessment of the K-Basin exterior walls and foundations determined they could withstand a 0.2*g* earthquake. Minor leakage could occur but would be slow. A recent assessment determined that the consequences of an earthquake for L- and P- Basins are less than those for K-Basin because K-Basin has the highest radionuclide inventory.	**Completed 3/95.** Soil stability assessments for K-, L-, and P-Basins are not required (Burke 1994). These assessments support seismic analyses of emergency cooling systems; however, the BIO (WSRC 1995c) determined that fuels stored in the basins do not require emergency cooling after a postulated Design-Basis Event resulting in basin draindown.
	SRS-21c: Depending on results of BIO completed by 11/94, development of accident mitigation procedures might be appropriate.	**Completed 7/96.** Accident mitigation procedures are not required because this accident would result in low consequences.
	SRS-21d: See Item 5,7,9a for reorientation of fuel.	**Completed 7/97.**
SRS-22: Potential vulnerability in buried fuel at SRS.	**SRS-22:** Fuel failure of a Mark V fuel assembly occurred in the R-Area disassembly basin in 1957. Over 7 years, the fissile materials in the failed fuel assembly completely oxidized. In 1964, the remains of the assembly were retrieved with no appreciable amount of fuel or fission products remaining in the assembly. All of the oxidized fuel near the assembly was removed using filters and deionizers and subsequently processed in RBOF. The fuel material is currently held in authorized basins and tanks. No further action is required.	**Completed 12/93.**

NOTE: This item was discussed in the Spent Fuel Working Group Report summary but not addressed in vulnerability development forms. Disassembly basins do not have high efficiency (confinement/negative pressure) ventilation systems.	**NOTE:** The K-Area Basis for Interim Operation addresses the existing facility and safety margins with respect to the need for airborne release containment. The results of the BIO show there is adequate safety margin without facility upgrades. Upgrades are not required for several reasons: - Over 40 years of operating experience with no events involving the spread of particulate radioactive contamination from the basins. - Little stored energy in the disassembly basins because of radioactive decay. - Radiation level in the disassembly basin area is low (<2 millirem/hour) and the contribution from airborne particulate matter is negligible.	**Completed 3/95.** Area BIO was approved by DOE; no further action is required (WSRC 1995c).

DNFSB 94-1 addressed vulnerabilities at several DOE sites, including the following SRS vulnerabilities concerning SNF and related solutions, tanks, and processing activities:

> Several large tanks in the F-Canyon at the Savannah River Site contain tens of thousands of gallons of solutions of plutonium and transplutonium isotopes. These tanks, their appendages, and vital support systems are old, subject to deterioration, prone to leakage, and they are not seismically qualified.
>
> Processing canyons and reactor basins at the Savannah River Site contain large amounts of deteriorating irradiated reactor fuel stored under conditions similar to those at the 603 Basin at INEL [Idaho National Engineering Laboratory].
>
> There are thousands of containers of plutonium-bearing liquids and solids at ... SRS .... Large quantities of plutonium solutions are stored in deteriorating tanks, piping, and plastic bottles.... It is well known that plutonium in contact with plastic can cause formation of hydrogen gas and pyrophoric plutonium compounds leading to a high probability of plutonium fires.
>
> The slow pace of remediation and additional delays in stabilizing materials might be accompanied by further deterioration of safety and unnecessary increased risks to workers and the public.

DOE accepted the Board's Recommendation on August 31, 1994, and issued *The Defense Nuclear Facilities Safety Board Recommendation 94-1 Implementation Plan* (DOE 1995b). Table B-3 summarizes the SRS vulnerabilities identified in DNFSB 94-1 and the associated commitments made by DOE in the Implementation Plan.

B.5 DNFSB January 1995 SRS Spent Fuel Vulnerability Assessment

In January 1995, members of the DNFSB staff assessed SRS progress toward resolving vulnerabilities associated with the storage of spent fuel. Their *SRS Spent Fuel Storage Trip Report,* January 23, 1995 (Burnfield 1995) stated that although DOE Headquarters had agreed to approach spent fuel vulnerability problems using a systems approach, they had been slow in implementing that approach and there was little evidence of SRS applying the approach to the spent fuel management project.

DNFSB members expressed concern that aggressive action was not being taken to resolve vulnerabilities related to improving the water chemistry of the basins. They also highlighted two new areas of concern related to RBOF.

The report acknowledged that Westinghouse Savannah River Company had initiated an aggressive program to ensure that the risks associated with these two new areas of concern were accurately quantified and were acceptable. However, the efforts for the two areas were not tied together and therefore could result in an inability to link the two hazards successfully.

The DNFSB did not formally submit these issues and the trip report to DOE. As a consequence, formal corrective actions were not developed nor were these issues entered into and tracked by the DOE Safety Issues Management System. However, Westinghouse Savannah River Company performed two nuclear criticality safety evaluations to address these issues: *Reactivity Effects of Tilting Fuel Assemblies and Bundles in RBOF* (Reed 1995) and *Credible Water Depth for Criticality Incidents*

Table B-3. Applicability of Defense Nuclear Facilities Safety Board Recommendation 94-1 to SRS.

Identified vulnerabilities	Implementation plan commitment	Status	
Sub-recommendation (3): That preparation be expedited to process dissolved plutonium and transplutonium isotopes in tanks in F-Canyon at the Savannah River Site into forms safer for interim storage. The Board considers this problem to be especially urgent.	A stabilization method for F-Canyon has been selected. Stabilization of plutonium solutions began in February 1995 and will be completed by January 1996. A conceptual design report for the stabilization of americium and curium solutions will be completed by December 1995. All americium and curium solutions will be stabilized by September 1998. Other solutions not specifically mentioned in this recommendation but addressed in this plan will be stabilized in accordance with the following schedule: • Plutonium-242 solution in H-Canyon by November 1997 • Highly enriched uranium solutions at SRS by December 1997 • Plutonium-239 solution in H-Canyon by February 2000 • Neptunium solutions in H-Canyon by December 2002	Stabilization of plutonium solution was completed in 4/96. The Conceptual Design Report for the stabilization of americium and curium solutions was completed in 11/95. The current schedule for americium/curium solution stabilization calls for completion by 9/02. This schedule may be rebaselined in 4/00. Stabilization of Plutonium-242 in H-Canyon was completed in 12/96. Highly enriched uranium solutions continue to be stored safely. The schedule for disposition of H-Canyon uranium solutions calls for stabilization by 12/03. This schedule may be rebaselined in 4/00. Stabilization of H-Canyon plutonium-239 solutions is forecast for completion in 2002, and stabilization of H-Canyon neptunium solutions is forecast for completion in 12/05. These schedules may be rebaselined in 4/00.	TC TC TC
Sub-recommendation (5): That preparation be expedited to process the containers of possibly unstable residues at the Rocky Flats Plant and to convert constituent plutonium to a form suitable for safe interim storage.	... Residues at other sites, not specifically addressed in this recommendation will be stabilized according to the following schedules: • Sand, slag, and crucibles at SRS by December 1997	Stabilization of sand, slag, and crucible at SRS began 10/97 and is forecast for completion by 7/98.	
Sub-recommendation (6): That preparations be expedited to process the deteriorating irradiated reactor fuel stored in basins at SRS into a form suitable for safe storage until an option for ultimate disposition is selected.	The method for stabilizing fuel and targets at SRS will be selected by July 1995 pursuant to the Interim Management of Nuclear Materials (IMNM) EIS and ROD. Fuel storage basin water chemistry upgrades will be completed by May 1996. Contingent on the outcome of the IMNM EIS, targets will be stabilized via dissolution by September 1996; fuel dissolution will be completed by November 1999. Stabilization of resultant uranium solutions will be completed by April 2000.	Stabilization of Mark 31 targets was completed in 1/97. Stabilization of Mark 16 and 22 fuel assemblies began 7/97 and is forecast for completion in 2001. Fuel storage basin water chemistry upgrades were completed in 5/96. HEU from fuel will be blended down to LEU on a schedule that supports transfer of the LEU to commercial industry.	
Sub-recommendation (8): That those facilities that may be needed for future handling and treatment of the materials in question be maintained in a usable state. Candidate facilities include, among others, F- and H-Canyons and FB- and HB-Lines at SRS, ...	Sufficient capabilities will be retained to maintain future handling, treatment and safe storage of the materials addressed in this plan. A discussion of facilities currently in use or planned for use is included in Section 2.6. The facilities section of the Integrated Program Plan will be prepared by December 1995.	The Integrated Facilities Plan (DOE 1995a) addressed the utilization of the F- and H- Canyons.	
Sub-recommendation (9): Expedited preparation to accomplish actions in items (3) through (8) above should take into account the need to meet the requirements for operational readiness in	Facilities will be started or restarted in accordance with DOE Order 5480.31. These restart and startup requirements will be taken into account in the development of the facilities.	Operational Readiness Reviews for restart of the following facilities have taken into account requirements of DOE Order 5480.31: FB-Line (complete 11/95); F-Canyon (complete 9/26); H-	

accordance with DOE Order 5480.31.		Canyon Dissolving (complete 7/97); HB-Line Dissolving (complete 3/98)

in RBOF (Reed 1996). Table B-4 summarizes the SRS vulnerabilities identified in the trip report and results of the safety evaluations.

B.6 DNFSB June 1995 SRS Spent Fuel Vulnerability Assessment

In June 1995, members of the DNFSB visited the SRS to review SNF activities related to the implementation of DNFSB Recommendation 94-1 (see Section B.4). The DNFSB Chairman formally transmitted the report to DOE and identified issues that were "... not being adequately considered in the evaluation of remediation alternatives." In addition, the report identified badly corroding foreign fuel in RBOF that DOE had categorized as stable (Conway 1995).

The corroding foreign fuel that DOE had categorized as stable was failed Taiwanese Research Reactor Fuel and Experimental Breeder Reactor slugs that were being stored in cans in RBOF. Although the damaged fuel was housed in protective cans, the cans were leaking and continued deterioration was likely. DOE responded to this concern by issuing the Interim Management of Nuclear Material EIS Record of Decision (60 FR 65300), which identified chemical processing as the preferred alternative for the at-risk foreign fuel.

In the *Draft Environmental Impact Statement, Interim Management of Nuclear Materials* (DOE 1995c), DOE identified Processing to Metal as the preferred alternative for the remediation of Mark-16 and -22 fuels. DNFSB became aware that DOE was considering dry storage of aluminum-clad fuel as an alternative to chemical processing. DOE issued the Final IMNM EIS (DOE 1995d) with No Action as the preferred alternative for these fuels to allow time for further consideration of dry storage. DNFSB expressed three specific concerns on the stabilization technologies for Mark-16 and -22 fuel. DOE's final decision was to identify Chemical Processing as the preferred alternative for this material, as recorded in the second IMNM Record of Decision (61 FR 6633).

Table B-5 summarizes the SRS vulnerabilities identified in the trip report.

Table B-4. Vulnerabilities identified in the January 1995 Defense Nuclear Facilities Safety Board Trip Report.

Identified vulnerabilities	Implementation plan commitment	Status
Some fuel in RBOF is stored vertically in racks, allowing the fuel to lean from top to bottom slightly, resulting in a violation of criticality safety requirements.	The DNFSB did not choose to submit these issues and this trip report formally to DOE. Consequently, formal corrective actions were not developed nor were these issues entered into and tracked by the DOE Safety Issues Management System.	Calculations documented in *Nuclear Criticality Safety Evaluation: Reactivity Effects of Tilting Fuel Assemblies and Bundles in RBOF* (Reed 1995) resulted in the conclusion that, "There are no current situations in RBOF Storage Basin #1 in which the configuration has been determined to be more reactive than a k_{eff} of 0.95, the Technical Standard limit."
The amount of water shielding was misidentified in the safety documentation.	The DNFSB did not choose to submit these issues and this trip report formally to DOE. Consequently, formal corrective actions were not developed nor were these issues entered into and tracked by the DOE Safety Issues Management System.	An analysis documented in *Nuclear Criticality Safety Evaluation: Credible Water Depth for Criticality Incidents in RBOF* (Reed 1996), concluded that, "Thus, there is no basis to define a NIM evacuation zone (region in which personnel can receive 12 rads or more as a result of a criticality incident) for the RBOF basins. Based on requirements of DOE Order 5480.24, a criticality

alarm system is not required for the RBOF basins.”

Table B-5. Vulnerabilities identified in the June 1995 Defense Nuclear Facilities Safety Board Trip Report.

Identified vulnerabilities	Implementation plan commitment	Status
1. Contrary to the Implementation Plan for Recommendation 94-1, it appears that dry storage is being considered as the preferred alternative to remediate Mark-16 and -22 fuel assemblies at the SRS. Although dry storage, as well as chemical separation, can achieve stable conditions, the following concerns could affect the decision to dry store this deteriorating fuel:		
1.a The requirements for dry storage of highly enriched aluminum-clad spent nuclear fuel have not been developed. This represents a large uncertainty in the time and effort required to achieve dry storage and a large uncertainty in the time during which continued wet storage will be required.	Formal corrective actions were not developed nor were these issues entered in and tracked by the DOE Safety Issues Management System.	The preferred alternative for remediation of these fuels was Blending Down to Low-Enriched Uranium, as recorded in the second IMNM Record of Decision (61 FR 6633).
1.b The waste acceptance criteria needed to transition dry-stored aluminum-clad spent nuclear fuel to a geologic repository have not been developed. This raises the possibility of having to rehandle, repackage, or even process this material in the future to meet storage requirements.	Formal corrective actions were not developed nor were these issues entered in and tracked by the DOE Safety Issues Management System.	The preferred alternative for remediation of these fuels was Blending Down to Low-Enriched Uranium, as recorded in the second IMNM Record of Decision (61 FR 6633).
1.c Lengthy delays needed to implement dry storage will extend by years the period of wet storage of the deteriorating spent fuel and allow continued corrosion. This will aggravate the problems of continued degradation, potential environmental insult, radiation exposure, and waste generation.	Formal corrective actions were not developed nor were these issues entered in and tracked by the DOE Safety Issues Management System.	DOE responded to these concerns by changing the preferred alternative for remediation of these fuels to Blending Down to Low-Enriched Uranium, as recorded in the second IMNM Record of Decision (61 FR 6633).
2. In addition, corroding spent fuel in RBOF is releasing more than twice the amount of fission products to the basin water than the corroding Mark-31 targets are releasing to the L-Basin. This significant corrosion is contaminating the facility, generating significant waste, and contributing to personnel exposure. Surprisingly, DOE plans to keep the current inventory of fuel at RBOF in wet storage for the next 10 years. A more urgent response is merited.	Formal corrective actions were not developed nor were these issues entered in and tracked by the DOE Safety Issues Management System.	DOE elected to stabilize all fuel in RBOF with potential leakage. (First Record of Decision [60 FR 65300] and fourth Record of Decision [62 FR 17790].)

B.7 DNFSB 1996 SRS Spent Fuel Handling Assessment

In August 1996, members of the DNFSB assessed SRS spent fuel handling and processing operations. The Board noted that the transfer of spent fuel requires moving massive casks in spent fuel storage basins where a cask drop could cause structural damage and significant water inventory loss. Their Trip Report (Conway 1996) reported the following concerns associated with stabilization operations and the retrieval of spent fuel from the K, L, and P Basins:

1. There is no assurance that makeup water will be available after a design-basis accident.
2. The crane rope is corroded, and the fatigue life of some cranes is not known.
3. A qualified rigger is not present during critical cask lifts.
4. Although fuel is being removed from the basins, significant quantities of activated scrap metal will remain.

DOE first responded to these concerns in a letter to the Board dated November 21, 1996 (Alm 1996b). DOE provided a more detailed response in a letter dated December 13, 1996 (Alm 1996c). Table B-6 summarizes the SRS SNF transfer vulnerabilities identified in the 1996 Trip Report and the associated DOE responses and commitments (Potvin 1997).

Table B-6. Vulnerabilities identified in the August 1996 Defense Nuclear Facilities Safety Board Trip Report.

Identified vulnerabilities	Implementation plan commitment	Status
There is no assurance that makeup water will be available if a cask drop or seismic event should cause a leak. Basin water is supposed to be replaced by raw untreated water from the Emergency Service Water system; however, this line is not tested regularly, it has not been used for more than a year, and it is not seismically qualified.	Studies indicate that dropping the 63.5 metric-ton (70-ton) cask, as analyzed in the recently issued Basis for Interim Operation for L-Reactor, could potentially result in a crack with a maximum leak of 397 liters (105 gallons) per minute. At this rate, operators would have a minimum of 6 days to implement mitigative actions before radiation levels began to increase, at which point all workers in the vicinity would be evacuated. In such an event, operators could respond by implementing various procedures using available systems to restore basin water levels. Although detailed preplanned and demonstrated emergency capabilities are not required for accident scenarios that would allow adequate time for facility workers to respond, an integrated facility response to a basin leak is being developed. This response, to be completed in the second quarter of FY 97, will consist of a combination of operational procedures and engineering response plans with the objective of mitigating basin leakage.	A letter to the DOE-SR DNFSB Liaison (Voss 1997a) identified plans and procedures developed in response to this vulnerability.
A qualified rigger is not present during fuel cask lifts. Fuel cask lifts are critical and preengineered. However, a crane operator, who has only Incidental Rigger Training, performs both the rigging and crane movement. This seems to contradict the SRS Hoisting and Rigging Manual, which states that a rigger shall ensure (1) the rigging equipment has the required capacity and is in good condition, (2) the rigging equipment is per procedure, and (3) the load path is clear.	Preengineered lifts are established for routine, repetitive lifting jobs such as cask movement. For these lifts, established procedures define the rigging equipment and the process used. These cask-handling procedures are reviewed by fully qualified Site rigging personnel. Facility operators attend "incidental rigger/operator" training. The "incidental rigger" is trained to ensure the rigging equipment has the required capacity and is in good condition, the rigging equipment is utilized per procedures, and the load path is clear. These qualifications are appropriate for the routine preengineered operations conducted in the facility.	This commitment is complete (WSRC 1997b).
Corrosion is evident along the entire length of the K-Basin cask crane's wire rope and the fatigue life of basin cranes is not known. DNFSB staff were not able to view the L-Basin crane rope, but were told its condition is similar. ASME B30.2-1990 identifies excessive corrosion on wire rope as a hazard. WSRC stated that the rope is adequate based on visual inspection by site riggers. However, as noted in the Construction Safety Association of Ontario's Rigging Manual, visual inspection gives a poor indication of the extent of degradation since corrosion often begins inside the rope. A more rigorous inspection includes examining	SRS has a comprehensive crane inspection program, which is based on a compilation of various national and international codes and standards. The quarterly and annual crane inspections are performed in accordance with *Overhead and Gantry Cranes*, ASME B30.2, Chapter 2-2. The wire ropes are inspected on a monthly frequency pursuant to "Overhead and Gantry Cranes," 29 CFR 1910.179(m), and inspection criteria in accordance with ASME B30.2, Section 2-2.4. Crane inspection program documents for the K- and L-Basins and the Receiving Basin for Offsite Fuel to be provided to the DNFSB will describe inspection requirements, frequency and criteria, load tests, dye-penetrant tests, etc.	DOE responded to this as documented in a letter (Sidey 1997). On January 20, 1997, additional references were provided in response to Commitment 2 (Voss 1997b).

the rope

Table B-6. (continued).

Identified vulnerabilities	Implementation plan commitment	Status
core. If the core is corroded, the cranes' safety factor may be much less than WSRC believes.	As part of scoping stage for determining the need for a crane upgrade proposal for DOE, a representative from the crane manufacturer will perform a baseline inspection/evaluation of the 11-metric-ton (85-ton) crane and issue a report. All documentation will be transmitted within 60 days of the crane manufacturer's inspection.	
Although fuel is being removed from the basins, significant quantities of activated scrap metal will remain. Besides fuel, the basins store buckets of highly radioactive scrap metal. These buckets are suspended by rope and corroding wire cables. In contrast with the fuel, no plan of action has been formulated for retrieval of this material.	As part of efforts to enhance water quality and reduce hazards, WSRC has been pursuing removal of selected materials from the basins. Activities to date have focused on removal of the more serious hazards, including cadmium control rods, corroded fuel, and excess radioactive sources. Future efforts will be directed at the remaining materials such as irradiated metal and contaminated scrap. DOE has recently improved methods for disposing of waste at SRS. These are part of a waste certification program that ensures the identification of characteristics of all wastes to enable proper disposal. The RBOF Waste Certification Plan is in revision and will reflect the new waste disposal process. Implementation schedules will be based on resource and budget availability.	After discussions with DOE, it was determined that no deliverable was required to be forwarded to DNFSB concerning this issue.

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APPENDIX C

SPENT NUCLEAR FUEL BACKGROUND AND INVENTORY

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APPENDIX C. SPENT NUCLEAR FUEL BACKGROUND AND INVENTORY

C.1 Background

C.1.1 GENERAL CHARACTERISTICS

In nuclear reactors a process occurs known as the fission chain reaction, in which atomic nuclei in reactor fuel respond to collisions with neutrons by splitting into two major fragments and two or three neutrons. The neutrons can interact with other fuel nuclei, thereby continuing the chain reaction.

In comparison to a chemical reaction involving the same mass, a nuclear reaction releases a large amount of energy, mostly the kinetic energy of the fission fragments and neutrons and the subsequent radioactive decay of the fission fragments (fission products). This energy makes nuclear fission an attractive source of energy for commercial power producers. DOE operated its production reactors principally because the neutrons caused nonfission nuclear reactions of interest to national defense (i.e., isotopic transmutation). Research reactors use the fission process to produce medical isotopes or for other research purposes.

Nuclear fuel must contain atoms that can be fissioned (called fission atoms). Fission atoms are fissioned by low-energy (thermal) neutrons. Therefore, to maintain the chain reaction, the high energy, fast neutrons produced by fission must be slowed to low-energy, thermal neutrons. The process for slowing down the neutrons is called moderation: water, graphite, and heavy water are used as moderators.

Uranium-235 is the fissile atom used most often for nuclear fuel; however, other fissile materials (uranium-233, plutonium-239, and plutonium-241) can be used in nuclear reactors. Uranium-235 represents only about 0.7 percent of the atoms of natural uranium, which is primarily uranium-238. Therefore, many reactors use fuel that has an enriched uranium-235 content. Commercial power reactors typically use fuels enriched to approximately 2 to 4 percent. Non-commercial reactors, depending on their purpose, use fuel enriched to as much as 93 percent uranium-235. Low-enriched uranium (LEU) has an enrichment below 20 percent; highly enriched uranium (HEU) is enriched 20 percent or higher. The fuels discussed in this EIS are primarily highly enriched uranium fuels.

The uranium in nuclear fuels generally is clad with a metal to protect it from chemical reactions with the moderator water and to prevent the release of fission products to the water. Zirconium, stainless steel, and aluminum are common cladding materials. Most of the SNF analyzed in this EIS (about 48 metric tons heavy metal [MTHM]) is aluminum-clad; the remainder is clad with stainless steel or zirconium.

Inside the cladding, the fuel is often in the form of a ceramic, an alloy that combines uranium with aluminum, metallic uranium, or a uranium oxide or silicide. The fuel can be assembled as parallel plates, concentric tubes, bundles of rods or pins, or other designs. Each assembly has mounting and lifting hardware, structures to direct coolant and moderator flow, and in some cases the capability to install neutron absorbing material and instrumentation. Usually a number of fuel assemblies make up a complete reactor core.

Spent nuclear fuel (SNF) is fuel that has been irradiated in a reactor and contains fissile atoms and fission products. SNF management must consider four fuel characteristics: radiation fields, heat generation, criticality, and chemical stability (corrosion resistance). As the fuel is irradiated in a reactor, much of the uranium is burned, resulting in the production of fission products. These fission products are radioactive; that is, they do not undergo fission but they radiate energy and transmute to other elements. SNF has very high radiation fields, especially for a

EC

period of time immediately after it is removed from the reactor. After a period of decay, as the short-lived fission products decay away, the radiation fields decrease, but the fuel is still highly radioactive and requires management for many years.

The heat from the radioactive decay of fission products (decay heat) can produce very high temperatures, requiring fuel recently removed from a reactor to be placed in underwater storage for cooling. Without active cooling, the fuel could overheat and melt or damage the cladding. After a sufficient cooling time that depends on the burnup of the fuel and its composition, fuel assemblies can be stored dry. Dry fuel storage technologies must be designed to release residual decay heat.

Long-term storage of SNF in water can lead to corrosion of the fuel cladding. Careful control of water chemistry can reduce the rate of corrosion. Aluminum-clad fuels, which are considered in this EIS, are more prone to corrosion in water than are stainless-steel or zirconium-clad fuels.

Most SNF could undergo a fission chain reaction. However, the fuel density, geometry, temperature, and moderation must support fission, or the chain reaction would not occur because too many neutrons would be absorbed or otherwise lost.

When a reactor is producing enough neutrons to support a chain reaction, it is termed “critical.” Criticality occurs when fissile material begins to undergo a chain reaction. SNF management must consider the potential of the fuel to create an unwanted criticality.

SNF can be chemically processed to recover transmitted isotopes for defense or commercial purposes and the fissile and fertile material for conversion into more nuclear fuel.

C.1.2 RECENT SPENT FUEL MANAGEMENT ACTIONS

In 1992, DOE decided to phase out defense-related SNF processing. Subsequently, the Department began to establish programs to manage DOE SNF that were no longer based on the production of strategic nuclear material. DOE identified the initial components of this plan in the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995a) (hereafter referred to as the Programmatic SNF EIS). The Record of Decision for this environmental impact statement (EIS) (60 FR 28680) stated in part that DOE would consolidate the management of its aluminum-clad SNF at the Savannah River Site (SRS) and would consolidate nonaluminum-clad fuels at the Idaho National Engineering and Environmental Laboratory. As a result, about 20 MTHM of stainless-steel and zirconium-clad SNF stored at SRS was designated for shipment to the Idaho National Engineering and Environmental Laboratory for management. In addition, DOE decided to ship about 10 MTHM of aluminum-clad SNF to SRS from domestic, and DOE research reactors, and the Idaho National Engineering and Environmental Laboratory.

However, in the Programmatic SNF EIS Record of Decision DOE made no decisions on the technologies it would apply to the management of SNF at the designated storage sites. The Record of Decision stated that the selection of SNF stabilization technologies and the preparation of SNF for ultimate disposition would be the subject of site-specific and fuel-type-specific evaluations prepared in accordance with the National Environmental Policy Act and tiered from the Programmatic SNF EIS (DOE 1995a).

In October 1995, DOE assessed the environmental impacts of stabilizing certain nuclear materials at SRS that presented potential environment, safety, and health vulnerabilities (DOE 1995b). The material evaluated by DOE included SRS production reactor SNF stored in the reactor disassembly basins and research reactor SNF stored in the Receiving Basin for Offsite Fuel. The Department decided to stabilize SNF that presented potential environmental, safety, and health vulnerabilities by processing the material through the existing chemical separations facilities at SRS. Under these decisions (60 FR 65300, 61 FR 6633, and 62 FR 17790), about 175 MTHM of the approximately 195 MTHM of SNF at SRS will be stabilized. After stabilization, the resulting material will be treated and managed so that it is acceptable for permanent disposition once those decisions are made. DOE concluded the remaining material, all of which was stored in the Receiving Basin for Offsite Fuel, was stable and could remain as is for several years pending disposition decisions. In addition, DOE decided some of the stable material might have programmatic value, that is, be of use to future DOE missions. Mark-18 targets stored in the Receiving Basin for Offsite Fuel could be shipped to other DOE sites for programmatic uses, including irradiation for transuranium isotope production (primarily for National Aeronautics and Space Administration use) and defense stockpile stewardship activities.

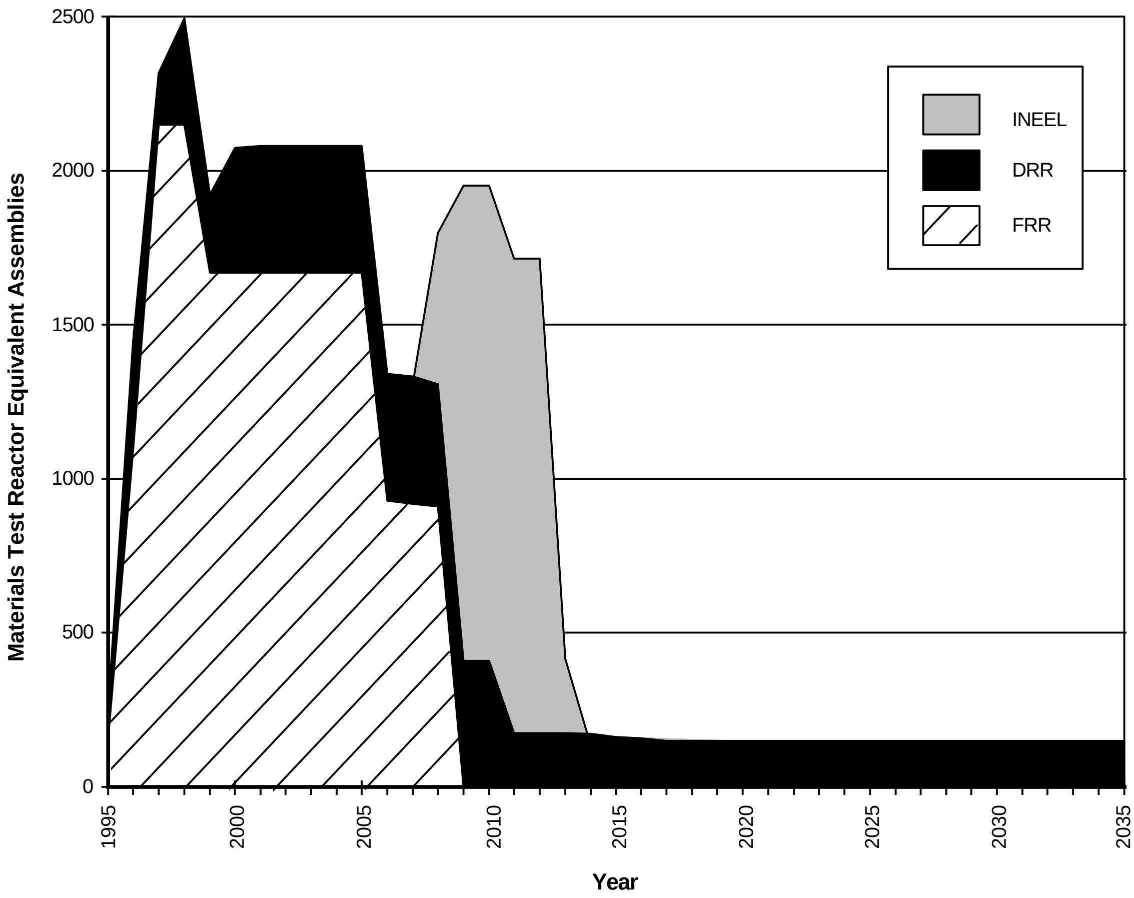
In May 1996 DOE issued its Record of Decision (61 FR 25092) for the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor [FRR] Spent Nuclear Fuel* (DOE 1996a) (hereafter referred to as the FRR EIS). The Department decided to accept and manage foreign research reactor SNF that contains uranium enriched in the United States. In keeping with its 1995 programmatic decision (60 FR 28680), DOE decided it would manage the aluminum-clad portion of the foreign research reactor SNF, about 18 MTHM, at SRS. Under the foreign research reactor receipt program, shipments from foreign reactors to SRS began in September 1996 and could continue for as long as 13 years. At present, SRS is receiving this fuel in either the Receiving Basin for Offsite Fuel or the L-Reactor Disassembly Basin. Figure C-1 shows projected receipts of aluminum-clad SNF at SRS from foreign and domestic sources, based on 1996 estimates. Because some countries may choose not to participate in the return of foreign SNF, the amount of aluminum-based foreign SNF to be managed at SRS may be less.

The May 1996 decision to accept foreign research reactor SNF for management in the United States (61 FR 25092) stated that DOE would issue a separate Record of Decision, after appropriate environmental reviews, to announce its plans for the management of such fuel. The Department committed to the aggressive pursuit of one or more new packaging or non-processing technologies that would put foreign research reactor SNF in a form or container suitable for disposal in a geologic repository. DOE also committed to place foreign research reactor SNF in dry storage at SRS (after required treatment or packaging) pending offsite storage or disposal. DOE also stated that if a new treatment technology was not ready for implementation by 2000, DOE would consider the chemical separation of some foreign reactor SNF that would blend the material down to low-enriched uranium in F Canyon at SRS. DOE might then place it under International Atomic Energy Agency safeguards.

C.2 Inventory

C.2.1 PHYSICAL INVENTORY

As Figure C-1 indicates, most SNF receipts would occur before 2015; however, SRS will continue to receive small amounts over the entire period of analysis (until 2035). There is great variety in the SNF that SRS must manage over the next 40 years. Therefore, DOE has categorized the SNF into six groups to facilitate analysis. The SNF in each category should receive nearly identical management. The basis for the categorization was often the size of the fuel in relation to packaging; however, other considerations were included such as physical characteristics, chemical characteristics, and radionuclide content. For example, one category includes all SNF in powder form. The following subsections describe the six fuel groups and list the SNF inventory associated associated with each group. Receipts per year will be aproximately 150 Materials Test M4-13



INEEL = Idaho National Engineering and Environmental Laboratory
DRR = Domestic Research Reactor
FRR = Foreign Research Reactor
Source: DOE 1996b.

Figure C-1. Projected receipts of SNF at the Savannah River Site.

Reactor-like Elements from domestic reactors and 12 High Flux Isotope Reactor assemblies from Oak Ridge.

C.2.1.1 Group A: Uranium and Thorium Metal Fuels

Group A consists mostly of chemically reactive uranium and thorium metal fuels. Many of the fuel elements are declad, and much of this group consists of depleted or natural uranium. As indicated in Table C-1, Group A fuels consist of four fuel types. The Experimental Breeder Reactor-II Blanket Fuels have been declad and the depleted uranium slugs placed in aluminum cans. The Advanced Reactivity Measurement Facility (ARMF) Core Filter Block is a 6 × 6 × 24-inch (15.2 × 15.2 × 61-centimeter) block of depleted uranium. The Sodium Reactor Experiment fuel consists of declad thorium metal placed in 3.5-inch (8.9-centimeter) diameter by 110-inch (279-centimeter) long cans. The Mark-42 targets are unirradiated tubes of plutonium oxide in an aluminum matrix approximately 3.7 inches (9.4 centimeters) in diameter and 168 inches (426 centimeters) long.

C.2.1.2 Group B: Materials Test Reactor-Like Fuels

Group B is comprised mostly of Materials Test Reactor fuels, as described in Section 1.5 and

Figure 1-3, plus a few other fuels of similar size and composition. Table C-2 lists the Group B inventory

Table C-1. Inventory of Group A SNF.

Name	Items	Units	Location
Experimental Breeder Reactor Blankets	59	Cans	SRS Wet Basins[a]
ARMF Core Filter Block	1	Filter	INEEL
Sodium Reactor Experiment	36	Cans	SRS Wet Basins
Mark-42	16	Bundles	SRS Wet Basins

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

ARMF = Advanced Reactivity Measurement Facility

INEEL = Idaho National Engineering and Environmental Laboratory

C.2.1.3 Group C: HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging

Group C fuels are similar in composition to Group B fuels in that they are aluminum-clad, highly enriched uranium and low enriched uranium oxides and silicides, but their size or shape precludes packaging without resizing or special packaging considerations. Some of the Group C SNF is smaller in diameter and longer than Group B fuels. Other fuel in this group is larger than Group B fuels in both diameter and length and often comes in odd shapes such as 0.5-by-0.9-meter (1.5-by-3-foot) cylinders or spheres with a diameter of 74 centimeters (29 inches). Table C-3 lists Group C inventory.

C.2.1.4 Group D: Loose Uranium Oxide in Cans

Group D fuels consist of loose uranium oxide and fission products in aluminum cans. Table C-4 lists the Group D inventory.

The Sterling Forest Oxide material in this fuel group is a residue of highly enriched uranium, plutonium, fission products, mixed oxides (chromium, nickel, iron, barium), barium acetate, and barium nitrate which resulted from the production of medical isotopes (primarily molybdenum 99). The material was plated on the inside of stainless steel tubes when it was irradiated. The material was then removed from the tubes with an acid flush and the uranium was recovered from a nitrate-sulfate solution, after eliminating the sulfate by precipitating with barium acetate and filtering. The filtrate was evaporated and pyrolyzed at 300ºC to an oxide form in an aluminum can. The can was sealed and shipped to SRS where it was placed into storage in the Receiving Basin for Offsite Fuels. Both the can and the oxide powder it contains are readily dissolved in acid.

The other items in this fuel group are liquid targets that DOE assumes would be converted to oxide prior to shipment to SRS.

C.2.1.5 Group E: Higher Actinide Targets

Group E contains target materials used to generate radionuclides with atomic numbers beyond that of uranium. The targets are placed in nuclear reactors and irradiated with neutrons, which causes nonfission nuclear reactions. These targets are aluminum-clad plutonium oxide that contain significant quantities of americium and curium, which react under neutron irradiation to produce even higher atomic number elements such as californium. Table C-5 lists the Group E inventory.

C.2.1.6 Group F: Non-aluminum Clad Fuels

Group F comprises the large variety of non-aluminum-clad SNF at SRS that DOE must ship to the Idaho National Engineering and Environmental Laboratory under the Record of Decision for the Programmatic SNF EIS (DOE 1995a). Table C-6 lists the Group F inventory.

C.2.2 RADIONUCLIDE INVENTORY

The six SNF groups that DOE would manage at SRS possess diverse chemical, physical, and radiological characteristics. There is also diversity within any single fuel group. In the absence of detailed radionuclide characterization of the fuel, DOE has simplified the analyses for this EIS by developing an analytical construction called a Reference Fuel Assembly (Garrett et al. 1995). The Reference Fuel Assembly is used as a standard reference for scaling fuel group characteristics. This assembly is a composite of depleted uranium, highly enriched uranium, and special target radionuclides.

To determine the radionuclide inventories of each fuel group, DOE calculated the ratio of radioactivity of each nuclide in the Reference Fuel Assembly to the fissile mass of the Reference Fuel Assembly; multiplied the resulting ratios by the fissile mass of the fuel groups to obtain nuclide-specific inventories for each fuel group. DOE performed an identical calculation based on total heavy metal mass rather than fissile mass of each fuel group. DOE conservatively used the calculation (fissile mass ratio or total heavy metal mass ratio) that yielded the largest value of each radionuclide to calculate the inventory of each radionuclide for the fuel group. Scaling by fissile mass is important because the fission products potentially produce most of the radiological impacts. Scaling by heavy metal mass is important because heavy metal mass is an indicator of processing time and provides appropriate representation of Group A fuels which contain little fissile mass. Table C-7 lists the results of these calculations.

Table C-2. Inventory of Group B SNF.

Name	Items	Units	Location
FRR MTR	10,812	Assemblies	Argentina, Australia, Austria, Brazil, Canada, Chile, Colombia, France, Germany, Greece, Indonesia, Iran, Israel, Italy, Japan, Netherlands, Pakistan, Peru, Philippines, Portugal, South Africa, Spain, Sweden, Switzerland, Taiwan, Thailand, Turkey, United Kingdom, Uruguay, Venezuela
DRR MTR	11,799	Assemblies	LANL, INEEL, ORNL, Brookhaven National Laboratory, Massachusetts Institute of Technology, Georgia Institute of Technology, Iowa State University, University of Massachusetts-Lowell, University of Michigan, Missouri University, Purdue University, Rhode Island Nuclear Center, University of Virginia, Worcester, National Institute of Standards and Technology
MTR	Approximately 1,100	Assemblies	SRS Wet Basins[a]
Cylindrical MTR	145	Assemblies	Japan
Box MTR	28	Assemblies	Japan
Tube MTR	4,077	Assemblies	Australia, Belgium, Denmark, Germany
Missouri University Research Reactor	224	Assemblies	SRS Wet Basins, MURR, INEEL
Advanced Test Reactor	3,132	Assemblies	INEEL
Advanced Reactivity Measurement Facility	67	Assemblies	INEEL
University of Washington	26	Bundles	INEEL
Advanced Reactivity Measurement Facility Plate	15	Plates	INEEL
Sterling Forest Fuel	200	Assemblies	SRS Wet Basins

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a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

b. This value changes with FRR and DRR ongoing receipts. Some double counting with FRR and DRR entries exists.

LANL = Los Alamos National Laboratory
ORNL = Oak Ridge National Laboratory
MTR = Materials Test Reactor
FRR = Foreign Research Reactor
MURR = Missouri University Research Reactor
DRR = Domestic Research Reactor
INEEL = Idaho National Engineering and Environmental Laboratory

Table C-3. Inventory of Group C SNF.

	Name	Items	Units	Location
	Mark-14	1	Can	SRS Wet Basins[a]
	Oak Ridge Research Reactor	165	Assemblies	SRS Wet Basins
	HWCTR	1	Can	SRS Wet Basins
	Pin bundle	12	Bundles	Canada, Jamaica
	Pin cluster	2,792	Clusters	Canada, South Korea
	ZPTR	45	Assemblies	Cornell University
	ZPR	17	Assemblies	Manhattan University
	OSR	24	Assemblies	Ohio State
	Argonaut	50	Assemblies	Florida
	Reactor a-Haut Flux	90	Assemblies	SRS Wet Basins, France
	High Flux Isotope Reactor	540	Assemblies	ORNL
	High Flux Isotope Reactor	1	Can	SRS Wet Basins
TC	BSR	32	Assemblies	ORNL/SRS
	Tower Shielding Reactor	1	Element	ORNL
	Tower Shielding Reactor	2	Cans	ORNL
TC	Sandia Pulse Reactor	43	Assemblies	Sandia National Laboratories
	Oak Ridge Reactor	9	Cans	ORNL/SRS

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a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.
BSR = Bulk Shielding Reactor
OSR = Ohio State Reactor
ZPR = Zero Power Reactor
ZPTR = Zero Power Test Reactor
ORNL = Oak Ridge National Laboratory
HWCTR = Heavy Water Components Test Reactor

Table C-4. Inventory of Group D SNF.

Name	Items	Units	Location
Sterling Forest Oxide	676	Cans	SRS Wet Basins[a]
Other non-MTR targets	6,750	Cans	Canada, Belgium, Argentina, Indonesia

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a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.
MTR = Materials Test Reactor

Table C-5. Inventory of Group E SNF.

Name	Items	Units	Location
Mark-18	65	Assemblies	SRS Wet Basins[a]
Mark-51	60	Slugs	SRS Wet Basins
Other	114	Slugs	SRS Wet Basins

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a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

Table C-6. Inventory of Group F SNF.

Name	Items	Units	Current Location
Carolinas-Virginia Tube Reactor	3	Bundles	SRS Wet Basins[a]
Dresden	24	Sleeves	SRS Wet Basins
Dresden	6	Cans	SRS Wet Basins
Elk River Reactor	38	Bundles	SRS Wet Basins
LWR Samples	5	Cans	SRS Wet Basins
H. B. Robinson	1	Can	SRS Wet Basins
Saxton	13	Bundles	SRS Wet Basins
Saxton	3	Cans	SRS Wet Basins
Saxton	3	Test Tubes	SRS Wet Basins
Vallecitos	2	Bundles	SRS Wet Basins
Babcock & Wilcox Scrap	1	Can	SRS Wet Basins
EBR-II (ANL-MXOX)	1	Cans	SRS Wet Basins
EBWR	6	Cans	SRS Wet Basins
EBWR	4	Bundles	SRS Wet Basins
EBWR	288	Assemblies	SRS Wet Basins
EPRI	1	Can	SRS Wet Basins
GCRE	6	Cans	SRS Wet Basins
GCRE	66	Assemblies	SRS Wet Basins
HWCTR	34	Slugs	SRS Wet Basins
HWCTR	87	Cans	SRS Wet Basins
HWCTR	57	Assemblies	SRS Wet Basins
HWCTR	22	Bundles	SRS Wet Basins
HWCTR	9	Test Tubes	SRS Wet Basins
HTRE	13	Cans	SRS Wet Basins
ML-1	68	Assemblies	SRS Wet Basins
ORNL S1W-1 rods	3	Cans	SRS Wet Basins
ORNL Mixed Oxide (BW-1)	1	Can	SRS Wet Basins
Shippingport	127	Pins	SRS Wet Basins
SPERT-3	3	Cans	SRS Wet Basins
Sodium Reactor Experiment (Carbide)	1	Can	SRS Wet Basins
CANDU	3	Cans	SRS Wet Basins
CANDU	56	Rods	SRS Wet Basins

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a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

EBR = Experimental Breeder Reactor
HWCTR = Heavy Water Components Test Reactor
CANDU = Canadian Deuterium-Uranium Reactor
LWR = Light Water Reactor
EBWR = Experimental Boiling Water Reactor
ANL-MXOX = Argonne National Laboratory Mixed Oxide
ORNL = Oak Ridge National Laboratory
GCRE = Gas Cooled Reactor Experiment
HTRE = High Temperature Reactor Experiment
ML-1 = Mobile Low Power Plant No. 1
SPERT-3 = Special Power Excursion Test-3
EPRI = Electric Power Research Institute

Table C-7. Radionuclide inventories based on the Reference Fuel Assembly (curies).

	Reference Fuel	Fuel Group					
Nuclide[a]	Assembly	A	B	C	D	E	F
H-3	51.6	2540	144,000	46,000	9,090	112	9,780
Kr-85	1,050	51,700	2,920,000	935,000	185,000	2,270	199,000
Sr-89	49.2	2420	137000	43800	8670	107	9320
Sr-90	8,080	398,000	22,500,000	7,200,000	1,420,000	17,500	1,530,000
Y-90	8,080	398,000	22,500,000	7,200,000	1,420,000	17,500	1,530,000
Y-91	213	10,500	593,000	190,000	37,500	461	40,400
Zr-95	454	22,400	1,260,000	404,000	80,000	983	86,000
Nb-95	1,010	49,800	2,810,000	899,000	178,000	2,190	191,000
Nb-95m	3.37	166	9,390	3,000	594	7.30	639
Tc-99	1.03	50.7	2,870	917	181	2.23	195
Rh-103m	1.96	96.6	5,460	1,750	345	4.25	371
Rh-106	21,100	1,040,000	58,800,000	18,800,000	3,720,000	45,700	4,000,000
Ru-103	2.17	107	6,040	1,930	382	4.70	411
Ru-106	21,100	1,040,000	58,800,000	18,800,000	3,720,000	45,700	4,000,000
Ag-110	2.32	114	6,460	2,070	409	5.03	440
Ag-110m	174	8,570	485,000	155,000	30,700	377	33,000
Cd-113m	6.95	342	19,400	6,190	1,220	15.1	1,320
Sn-119m	3.93	194	10,900	3,500	693	8.51	745
Sn-123	14.5	714	40,400	12,900	2,560	31.4	2,750
Sb-125	870	42,900	2,420,000	775,000	153,000	1,880	165,000
Te-125m	212	10,400	590,000	189,000	37,400	459	40,200
Te-127	34.7	1,710	96,600	30,900	6,110	75.2	6,570
Te-127m	35.4	1,740	98,600	31,500	6,240	76.7	6,710
Te-129	0.0012	591	3.34	1.07	0.211	0.0026	0.227
Te-129m	0.00185	911	5.15	1.65	0.326	0.00401	0.351
Cs-134	10,300	507,000	28,700,000	9,170,000	1,810,000	22,300	1,950,000
Cs-137	9,280	457,000	25,800,000	8,260,000	1,640,000	20,100	1,760,000
Ba-137m	8,780	433,000	24,500,000	7,820,000	1,550,000	19,000	1,660,000
Ce-141	0.0646	3.18	180	57.5	11.4	0.140	12.2
Ce-144	47,800	2,350,000	133,000,000	42,600,000	8,420,000	104,000	9,060,000
Pr-144	47,800	2,350,000	133,000,000	42,600,000	8,420,000	104,000	9,060,000
Pr-144m	574	28,300	1,600,000	511,000	101,000	1,240	109,000
Pm-147	18,800	926,000	52,400,000	16,700,000	3,310,000	40,700	3,560,000
Pm-148m	0.00893	0.44	24.9	7.95	1.57	0.0193	1.69
Sm-151	69.4	3,420	193,000	61,800	12,200	150	13,100
Eu-154	727	35,800	2,020,000	647,000	128,000	1,570	138,000
Eu-155	381	18,800	1,060,000	339,000	67,100	825	72,200
Tl-208	8.46	417	23,600	7,530	1,490	18.3	1,600
Pb-209	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Pb-211	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Pb-212	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Bi-211	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Bi-212	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Bi-213	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Po-212	15.1	744	42,100	13,400	2,660	32.7	2,860
Po-213	0.00855	0.421	23.8	7.61	1.51	0.0185	1.62

Table C-7. (continued).

Nuclide[a]	Reference Fuel Assembly	Fuel Group A	B	C	D	E	F
Po-215	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Po-216	23.6	1,160	65,700	21,000	4,160	51.1	4,470
At-217	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Rn-219	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Rn-220	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Fr-221	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ra-223	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Ra-224	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Ra-225	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ac-225	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ac-227	0.0171	0.842	47.6	15.2	3.01	0.037	3.24
Th-227	0.0164	0.808	45.7	14.6	2.89	0.0355	3.11
Th-228	23.5	1160	65,500	20,900	4,140	50.9	4,450
Th-229	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Th-231	0.0114	0.562	31.8	10.2	2.01	0.0247	2.16
Th-232	0.0172	0.847	47.9	15.3	3.03	0.0373	3.26
Th-234	0.000216	0.0106	0.602	0.192	0.0381	0.000468	0.0409
Pa-231	0.228	11.2	635	203	40.2	0.494	43.2
Pa-233	0.0859	4.23	239	76.5	15.1	0.186	16.3
Pa-234m	0.000216	0.0106	0.602	0.192	0.0381	0.000468	0.0409
U-232	40.9	2,010	114,000	36,400	7,210	88.6	7,750
U-233	39.3	1,940	109,000	35,000	6,930	85.1	7,450
U-234	1.92	94.6	5,350	1,710	338	4.16	364
U-235	0.0114	0.562	31.8	10.2	2.01	0.0247	2.16
U-236	0.0329	1.62	91.6	29.3	5.80	0.0713	6.23
U-237	0.259	12.8	721	231	45.6	0.561	49.1
U-238	0.0842	4.15	235	75.0	14.8	0.182	16.0
Np-237	0.00881	0.434	24.5	7.85	1.55	0.02	1.67
Np-239	9.62	474	26,800	8,570	1,700	21	1,820
Pu-236	112	5,520	312,000	99,700	19,700	250	21,200
Pu-238	51.9	2,560	145,000	46,200	9,150	340	9,830
Pu-239	58	2,860	162,000	51,700	10,200	130	11,000
Pu-240	9,780	482,000	27,200,000	8,710,000	1,720,000	23,000	1,850,000
Pu-241	10,600	522,000	29,500,000	9,440,000	1,870,000	23,000	2,010,000
Am-241	51.7	2,550	144,000	46,000	9,110	450	9,800
Am-242	0.34	16.7	947	303	59.9	0.74	64.4
Am-242m	0.341	16.8	950	304	60.1	0.74	64.6
Am-243	9.62	474	26,800	8,570	1,700	21	1,820
Cm-242	490	24,100	1,360,000	436,000	86,300	1,100	92,800
Cm-243	4.9	241	13,600	4,360	863	11	928
Cm-244	2,750	135,000	7,660,000	2,450,000	485,000	18,000	521,000
Cm-246	0.215	10.6	599	191	37.9	150	40.7
Totals	231,000	11,400,000	644,000,000	206,000,000	40,700,000	520,000	43,800,000

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a. Refer to Table C-8 for the names of the elements.

Table C-8. Chemical symbols used in Table C-7 and the corresponding element names.

H-3 = tritium
Kr = krypton
Sr = strontium
Y = yttrium
Zr = zirconium
Nb = niobium
Tc = technetium
Rh = rhodium
Ru = ruthenium
Ag = silver
Cd = cadmium
Sn = tin
Sb = antimony
Te = tellurium
Cs = cesium
Ba = barium
Ce = cerium
Pr = praseodymium
Pm = promethium
Sm = samarium
Eu = europium
Tl = thallium
Pb = lead
Bi = bismuth
Po = polonium
At = astatine
Rn = radon
Fr = francium
Ra = radium
Ac = actinium
Th = thorium
Pa = protactinium
U = uranium
Np = neptunium
Pu = plutonium
Am = americium
Cm = curium

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APPENDIX C

SPENT NUCLEAR FUEL BACKGROUND AND INVENTORY

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APPENDIX C. SPENT NUCLEAR FUEL BACKGROUND AND INVENTORY

C.1 Background

C.1.1 GENERAL CHARACTERISTICS

In nuclear reactors a process occurs known as the fission chain reaction, in which atomic nuclei in reactor fuel respond to collisions with neutrons by splitting into two major fragments and two or three neutrons. The neutrons can interact with other fuel nuclei, thereby continuing the chain reaction.

In comparison to a chemical reaction involving the same mass, a nuclear reaction releases a large amount of energy, mostly the kinetic energy of the fission fragments and neutrons and the subsequent radioactive decay of the fission fragments (fission products). This energy makes nuclear fission an attractive source of energy for commercial power producers. DOE operated its production reactors principally because the neutrons caused nonfission nuclear reactions of interest to national defense (i.e., isotopic transmutation). Research reactors use the fission process to produce medical isotopes or for other research purposes.

Nuclear fuel must contain atoms that can be fissioned (called fission atoms). Fission atoms are fissioned by low-energy (thermal) neutrons. Therefore, to maintain the chain reaction, the high energy, fast neutrons produced by fission must be slowed to low-energy, thermal neutrons. The process for slowing down the neutrons is called moderation: water, graphite, and heavy water are used as moderators.

Uranium-235 is the fissile atom used most often for nuclear fuel; however, other fissile materials (uranium-233, plutonium-239, and plutonium-241) can be used in nuclear reactors. Uranium-235 represents only about 0.7 percent of the atoms of natural uranium, which is primarily uranium-238. Therefore, many reactors use fuel that has an enriched uranium-235 content. Commercial power reactors typically use fuels enriched to approximately 2 to 4 percent. Non-commercial reactors, depending on their purpose, use fuel enriched to as much as 93 percent uranium-235. Low-enriched uranium (LEU) has an enrichment below 20 percent; highly enriched uranium (HEU) is enriched 20 percent or higher. The fuels discussed in this EIS are primarily highly enriched uranium fuels.

The uranium in nuclear fuels generally is clad with a metal to protect it from chemical reactions with the moderator water and to prevent the release of fission products to the water. Zirconium, stainless steel, and aluminum are common cladding materials. Most of the SNF analyzed in this EIS (about 48 metric tons heavy metal [MTHM]) is aluminum-clad; the remainder is clad with stainless steel or zirconium.

Inside the cladding, the fuel is often in the form of a ceramic, an alloy that combines uranium with aluminum, metallic uranium, or a uranium oxide or silicide. The fuel can be assembled as parallel plates, concentric tubes, bundles of rods or pins, or other designs. Each assembly has mounting and lifting hardware, structures to direct coolant and moderator flow, and in some cases the capability to install neutron absorbing material and instrumentation. Usually a number of fuel assemblies make up a complete reactor core.

Spent nuclear fuel (SNF) is fuel that has been irradiated in a reactor and contains fissile atoms and fission products. SNF management must consider four fuel characteristics: radiation fields, heat generation, criticality, and chemical stability (corrosion resistance). As the fuel is irradiated in a reactor, much of the uranium is burned, resulting in the production of fission products. These fission products are radioactive; that is, they do not undergo fission but they radiate energy and transmute to other elements. SNF has very high radiation fields, especially for a

EC period of time immediately after it is removed from the reactor. After a period of decay, as the short-lived fission products decay away, the radiation fields decrease, but the fuel is still highly radioactive and requires management for many years.

The heat from the radioactive decay of fission products (decay heat) can produce very high temperatures, requiring fuel recently removed from a reactor to be placed in underwater storage for cooling. Without active cooling, the fuel could overheat and melt or damage the cladding. After a sufficient cooling time that depends on the burnup of the fuel and its composition, fuel assemblies can be stored dry. Dry fuel storage technologies must be designed to release residual decay heat.

Long-term storage of SNF in water can lead to corrosion of the fuel cladding. Careful control of water chemistry can reduce the rate of corrosion. Aluminum-clad fuels, which are considered in this EIS, are more prone to corrosion in water than are stainless-steel or zirconium-clad fuels.

Most SNF could undergo a fission chain reaction. However, the fuel density, geometry, temperature, and moderation must support fission, or the chain reaction would not occur because too many neutrons would be absorbed or otherwise lost.

When a reactor is producing enough neutrons to support a chain reaction, it is termed “critical.” Criticality occurs when fissile material begins to undergo a chain reaction. SNF management must consider the potential of the fuel to create an unwanted criticality.

SNF can be chemically processed to recover transmitted isotopes for defense or commercial purposes and the fissile and fertile material for conversion into more nuclear fuel.

C.1.2 RECENT SPENT FUEL MANAGEMENT ACTIONS

In 1992, DOE decided to phase out defense-related SNF processing. Subsequently, the Department began to establish programs to manage DOE SNF that were no longer based on the production of strategic nuclear material. DOE identified the initial components of this plan in the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995a) (hereafter referred to as the Programmatic SNF EIS). The Record of Decision for this environmental impact statement (EIS) (60 FR 28680) stated in part that DOE would consolidate the management of its aluminum-clad SNF at the Savannah River Site (SRS) and would consolidate nonaluminum-clad fuels at the Idaho National Engineering and Environmental Laboratory. As a result, about 20 MTHM of stainless-steel and zirconium-clad SNF stored at SRS was designated for shipment to the Idaho National Engineering and Environmental Laboratory for management. In addition, DOE decided to ship about 10 MTHM of aluminum-clad SNF to SRS from domestic, and DOE research reactors, and the Idaho National Engineering and Environmental Laboratory.

However, in the Programmatic SNF EIS Record of Decision DOE made no decisions on the technologies it would apply to the management of SNF at the designated storage sites. The Record of Decision stated that the selection of SNF stabilization technologies and the preparation of SNF for ultimate disposition would be the subject of site-specific and fuel-type-specific evaluations prepared in accordance with the National Environmental Policy Act and tiered from the Programmatic SNF EIS (DOE 1995a).

In October 1995, DOE assessed the environmental impacts of stabilizing certain nuclear materials at SRS that presented potential environment, safety, and health vulnerabilities (DOE 1995b). The material evaluated by DOE included SRS production reactor SNF stored in the reactor disassembly basins and research reactor SNF stored in the Receiving Basin for Offsite Fuel. The Department decided to stabilize SNF that presented potential environmental, safety, and health vulnerabilities by processing the material through the existing chemical separations facilities at SRS. Under these decisions (60 FR 65300, 61 FR 6633, and 62 FR 17790), about 175 MTHM of the approximately 195 MTHM of SNF at SRS will be stabilized. After stabilization, the resulting material will be treated and managed so that it is acceptable for permanent disposition once those decisions are made. DOE concluded the remaining material, all of which was stored in the Receiving Basin for Offsite Fuel, was stable and could remain as is for several years pending disposition decisions. In addition, DOE decided some of the stable material might have programmatic value, that is, be of use to future DOE missions. Mark-18 targets stored in the Receiving Basin for Offsite Fuel could be shipped to other DOE sites for programmatic uses, including irradiation for transuranium isotope production (primarily for National Aeronautics and Space Administration use) and defense stockpile stewardship activities.

In May 1996 DOE issued its Record of Decision (61 FR 25092) for the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor [FRR] Spent Nuclear Fuel* (DOE 1996a) (hereafter referred to as the FRR EIS). The Department decided to accept and manage foreign research reactor SNF that contains uranium enriched in the United States. In keeping with its 1995 programmatic decision (60 FR 28680), DOE decided it would manage the aluminum-clad portion of the foreign research reactor SNF, about 18 MTHM, at SRS. Under the foreign research reactor receipt program, shipments from foreign reactors to SRS began in September 1996 and could continue for as long as 13 years. At present, SRS is receiving this fuel in either the Receiving Basin for Offsite Fuel or the L-Reactor Disassembly Basin. Figure C-1 shows projected receipts of aluminum-clad SNF at SRS from foreign and domestic sources, based on 1996 estimates. Because some countries may choose not to participate in the return of foreign SNF, the amount of aluminum-based foreign SNF to be managed at SRS may be less.

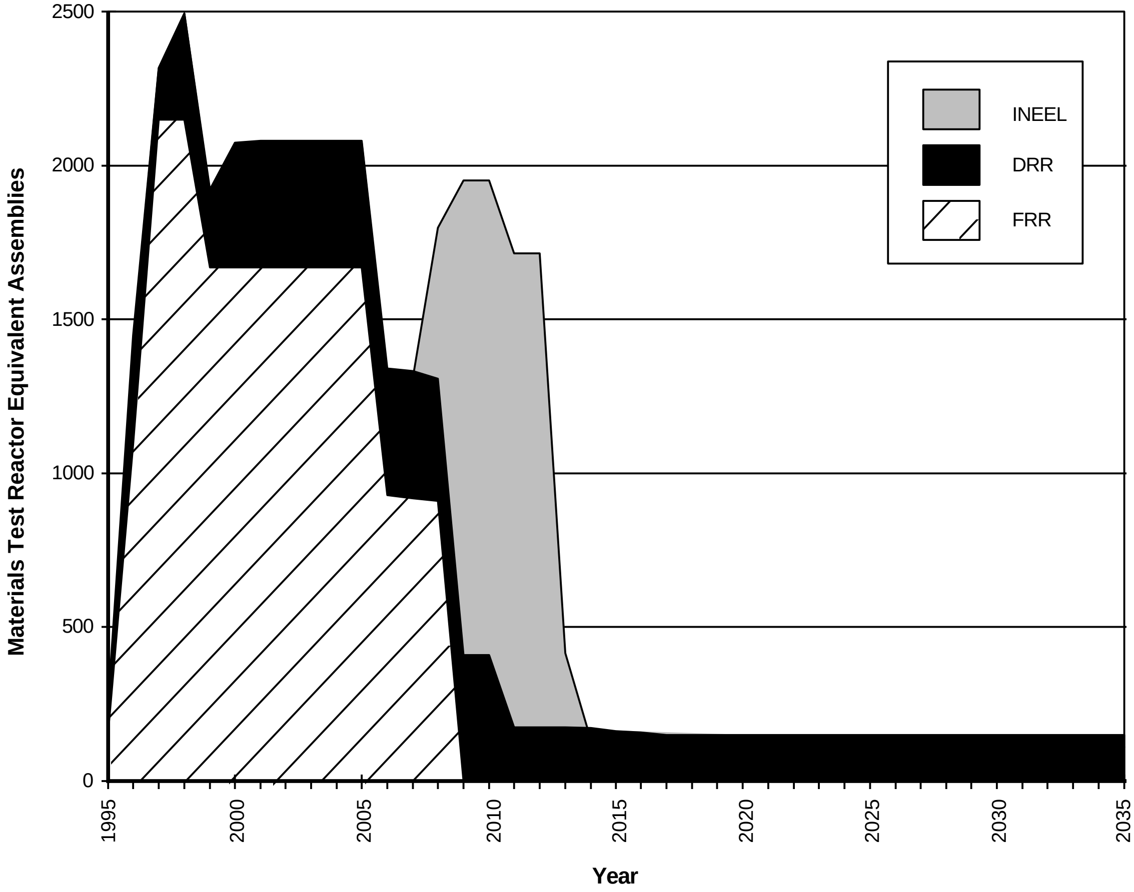
The May 1996 decision to accept foreign research reactor SNF for management in the United States (61 FR 25092) stated that DOE would issue a separate Record of Decision, after appropriate environmental reviews, to announce its plans for the management of such fuel. The Department committed to the aggressive pursuit of one or more new packaging or non-processing technologies that would put foreign research reactor SNF in a form or container suitable for disposal in a geologic repository. DOE also committed to place foreign research reactor SNF in dry storage at SRS (after required treatment or packaging) pending offsite storage or disposal. DOE also stated that if a new treatment technology was not ready for implementation by 2000, DOE would consider the chemical separation of some foreign reactor SNF that would blend the material down to low-enriched uranium in F Canyon at SRS. DOE might then place it under International Atomic Energy Agency safeguards.

C.2 Inventory

C.2.1 PHYSICAL INVENTORY

As Figure C-1 indicates, most SNF receipts would occur before 2015; however, SRS will continue to receive small amounts over the entire period of analysis (until 2035). There is great variety in the SNF that SRS must manage over the next 40 years. Therefore, DOE has categorized the SNF into six groups to facilitate analysis. The SNF in each category should receive nearly identical management. The basis for the categorization was often the size of the fuel in relation to packaging; however, other considerations were included such as physical characteristics, chemical characteristics, and radionuclide content. For example, one category includes all SNF in powder form. The following subsections describe the six fuel groups and list the SNF inventory associated associated with each group. Receipts per year will be aproximately 150 Materials Test

M4-13



INEEL = Idaho National Engineering and Environmental Laboratory
DRR = Domestic Research Reactor
FRR = Foreign Research Reactor
Source: DOE 1996b.

Figure C-1. Projected receipts of SNF at the Savannah River Site.

Reactor-like Elements from domestic reactors and 12 High Flux Isotope Reactor assemblies from Oak Ridge.

C.2.1.1 Group A: Uranium and Thorium Metal Fuels

Group A consists mostly of chemically reactive uranium and thorium metal fuels. Many of the fuel elements are declad, and much of this group consists of depleted or natural uranium. As indicated in Table C-1, Group A fuels consist of four fuel types. The Experimental Breeder Reactor-II Blanket Fuels have been declad and the depleted uranium slugs placed in aluminum cans. The Advanced Reactivity Measurement Facility (ARMF) Core Filter Block is a 6 × 6 × 24-inch (15.2 × 15.2 × 61-centimeter) block of depleted uranium. The Sodium Reactor Experiment fuel consists of declad thorium metal placed in 3.5-inch (8.9-centimeter) diameter by 110-inch (279-centimeter) long cans. The Mark-42 targets are unirradiated tubes of plutonium oxide in an aluminum matrix approximately 3.7 inches (9.4 centimeters) in diameter and 168 inches (426 centimeters) long.

C.2.1.2 Group B: Materials Test Reactor-Like Fuels

Group B is comprised mostly of Materials Test Reactor fuels, as described in Section 1.5 and

Figure 1-3, plus a few other fuels of similar size and composition. Table C-2 lists the Group B inventory

Table C-1. Inventory of Group A SNF.

Name	Items	Units	Location
Experimental Breeder Reactor Blankets	59	Cans	SRS Wet Basins[a]
ARMF Core Filter Block	1	Filter	INEEL
Sodium Reactor Experiment	36	Cans	SRS Wet Basins
Mark-42	16	Bundles	SRS Wet Basins

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

ARMF = Advanced Reactivity Measurement Facility

INEEL = Idaho National Engineering and Environmental Laboratory

C.2.1.3 Group C: HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging

Group C fuels are similar in composition to Group B fuels in that they are aluminum-clad, highly enriched uranium and low enriched uranium oxides and silicides, but their size or shape precludes packaging without resizing or special packaging considerations. Some of the Group C SNF is smaller in diameter and longer than Group B fuels. Other fuel in this group is larger than Group B fuels in both diameter and length and often comes in odd shapes such as 0.5-by-0.9-meter (1.5-by-3-foot) cylinders or spheres with a diameter of 74 centimeters (29 inches). Table C-3 lists Group C inventory.

C.2.1.4 Group D: Loose Uranium Oxide in Cans

Group D fuels consist of loose uranium oxide and fission products in aluminum cans. Table C-4 lists the Group D inventory.

The Sterling Forest Oxide material in this fuel group is a residue of highly enriched uranium, plutonium, fission products, mixed oxides (chromium, nickel, iron, barium), barium acetate, and barium nitrate which resulted from the production of medical isotopes (primarily molybdenum 99). The material was plated on the inside of stainless steel tubes when it was irradiated. The material was then removed from the tubes with an acid flush and the uranium was recovered from a nitrate-sulfate solution, after eliminating the sulfate by precipitating with barium acetate and filtering. The filtrate was evaporated and pyrolyzed at 300ºC to an oxide form in an aluminum can. The can was sealed and shipped to SRS where it was placed into storage in the Receiving Basin for Offsite Fuels. Both the can and the oxide powder it contains are readily dissolved in acid.

The other items in this fuel group are liquid targets that DOE assumes would be converted to oxide prior to shipment to SRS.

C.2.1.5 Group E: Higher Actinide Targets

Group E contains target materials used to generate radionuclides with atomic numbers beyond that of uranium. The targets are placed in nuclear reactors and irradiated with neutrons, which causes nonfission nuclear reactions. These targets are aluminum-clad plutonium oxide that contain significant quantities of americium and curium, which react under neutron irradiation to produce even higher atomic number elements such as californium. Table C-5 lists the Group E inventory.

C.2.1.6 Group F: Non-aluminum Clad Fuels

Group F comprises the large variety of non-aluminum-clad SNF at SRS that DOE must ship to the Idaho National Engineering and Environmental Laboratory under the Record of Decision for the Programmatic SNF EIS (DOE 1995a). Table C-6 lists the Group F inventory.

C.2.2 RADIONUCLIDE INVENTORY

The six SNF groups that DOE would manage at SRS possess diverse chemical, physical, and radiological characteristics. There is also diversity within any single fuel group. In the absence of detailed radionuclide characterization of the fuel, DOE has simplified the analyses for this EIS by developing an analytical construction called a Reference Fuel Assembly (Garrett et al. 1995). The Reference Fuel Assembly is used as a standard reference for scaling fuel group characteristics. This assembly is a composite of depleted uranium, highly enriched uranium, and special target radionuclides.

To determine the radionuclide inventories of each fuel group, DOE calculated the ratio of radioactivity of each nuclide in the Reference Fuel Assembly to the fissile mass of the Reference Fuel Assembly; multiplied the resulting ratios by the fissile mass of the fuel groups to obtain nuclide-specific inventories for each fuel group. DOE performed an identical calculation based on total heavy metal mass rather than fissile mass of each fuel group. DOE conservatively used the calculation (fissile mass ratio or total heavy metal mass ratio) that yielded the largest value of each radionuclide to calculate the inventory of each radionuclide for the fuel group. Scaling by fissile mass is important because the fission products potentially produce most of the radiological impacts. Scaling by heavy metal mass is important because heavy metal mass is an indicator of processing time and provides appropriate representation of Group A fuels which contain little fissile mass. Table C-7 lists the results of these calculations.

Table C-2. Inventory of Group B SNF.

Name	Items	Units	Location
FRR MTR	10,812	Assemblies	Argentina, Australia, Austria, Brazil, Canada, Chile, Colombia, France, Germany, Greece, Indonesia, Iran, Israel, Italy, Japan, Netherlands, Pakistan, Peru, Philippines, Portugal, South Africa, Spain, Sweden, Switzerland, Taiwan, Thailand, Turkey, United Kingdom, Uruguay, Venezuela
DRR MTR	11,799	Assemblies	LANL, INEEL, ORNL, Brookhaven National Laboratory, Massachusetts Institute of Technology, Georgia Institute of Technology, Iowa State University, University of Massachusetts-Lowell, University of Michigan, Missouri University, Purdue University, Rhode Island Nuclear Center, University of Virginia, Worcester, National Institute of Standards and Technology
MTR	Approximately 1,100	Assemblies	SRS Wet Basins[a]
Cylindrical MTR	145	Assemblies	Japan
Box MTR	28	Assemblies	Japan
Tube MTR	4,077	Assemblies	Australia, Belgium, Denmark, Germany
Missouri University Research Reactor	224	Assemblies	SRS Wet Basins, MURR, INEEL
Advanced Test Reactor	3,132	Assemblies	INEEL
Advanced Reactivity Measurement Facility	67	Assemblies	INEEL
University of Washington	26	Bundles	INEEL
Advanced Reactivity Measurement Facility Plate	15	Plates	INEEL
Sterling Forest Fuel	200	Assemblies	SRS Wet Basins

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a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

b. This value changes with FRR and DRR ongoing receipts. Some double counting with FRR and DRR entries exists.

LANL = Los Alamos National Laboratory
ORNL = Oak Ridge National Laboratory
MTR = Materials Test Reactor
FRR = Foreign Research Reactor
MURR = Missouri University Research Reactor
DRR = Domestic Research Reactor
INEEL = Idaho National Engineering and Environmental Laboratory

Table C-3. Inventory of Group C SNF.

Name	Items	Units	Location
Mark-14	1	Can	SRS Wet Basins[a]
Oak Ridge Research Reactor	165	Assemblies	SRS Wet Basins
HWCTR	1	Can	SRS Wet Basins
Pin bundle	12	Bundles	Canada, Jamaica
Pin cluster	2,792	Clusters	Canada, South Korea
ZPTR	45	Assemblies	Cornell University
ZPR	17	Assemblies	Manhattan University
OSR	24	Assemblies	Ohio State
Argonaut	50	Assemblies	Florida
Reactor a-Haut Flux	90	Assemblies	SRS Wet Basins, France
High Flux Isotope Reactor	540	Assemblies	ORNL
High Flux Isotope Reactor	1	Can	SRS Wet Basins
BSR	32	Assemblies	ORNL/SRS
Tower Shielding Reactor	1	Element	ORNL
Tower Shielding Reactor	2	Cans	ORNL
Sandia Pulse Reactor	43	Assemblies	Sandia National Laboratories
Oak Ridge Reactor	9	Cans	ORNL/SRS

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[a] Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

BSR = Bulk Shielding Reactor
OSR = Ohio State Reactor
ZPR = Zero Power Reactor
ZPTR = Zero Power Test Reactor
ORNL = Oak Ridge National Laboratory
HWCTR = Heavy Water Components Test Reactor

Table C-4. Inventory of Group D SNF.

Name	Items	Units	Location
Sterling Forest Oxide	676	Cans	SRS Wet Basins[a]
Other non-MTR targets	6,750	Cans	Canada, Belgium, Argentina, Indonesia

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[a] Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

MTR = Materials Test Reactor

Table C-5. Inventory of Group E SNF.

Name	Items	Units	Location
Mark-18	65	Assemblies	SRS Wet Basins[a]
Mark-51	60	Slugs	SRS Wet Basins
Other	114	Slugs	SRS Wet Basins

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a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

Table C-6. Inventory of Group F SNF.

Name	Items	Units	Current Location
Carolinas-Virginia Tube Reactor	3	Bundles	SRS Wet Basins[a]
Dresden	24	Sleeves	SRS Wet Basins
Dresden	6	Cans	SRS Wet Basins
Elk River Reactor	38	Bundles	SRS Wet Basins
LWR Samples	5	Cans	SRS Wet Basins
H. B. Robinson	1	Can	SRS Wet Basins
Saxton	13	Bundles	SRS Wet Basins
Saxton	3	Cans	SRS Wet Basins
Saxton	3	Test Tubes	SRS Wet Basins
Vallecitos	2	Bundles	SRS Wet Basins
Babcock & Wilcox Scrap	1	Can	SRS Wet Basins
EBR-II (ANL-MXOX)	1	Cans	SRS Wet Basins
EBWR	6	Cans	SRS Wet Basins
EBWR	4	Bundles	SRS Wet Basins
EBWR	288	Assemblies	SRS Wet Basins
EPRI	1	Can	SRS Wet Basins
GCRE	6	Cans	SRS Wet Basins
GCRE	66	Assemblies	SRS Wet Basins
HWCTR	34	Slugs	SRS Wet Basins
HWCTR	87	Cans	SRS Wet Basins
HWCTR	57	Assemblies	SRS Wet Basins
HWCTR	22	Bundles	SRS Wet Basins
HWCTR	9	Test Tubes	SRS Wet Basins
HTRE	13	Cans	SRS Wet Basins
ML-1	68	Assemblies	SRS Wet Basins
ORNL S1W-1 rods	3	Cans	SRS Wet Basins
ORNL Mixed Oxide (BW-1)	1	Can	SRS Wet Basins
Shippingport	127	Pins	SRS Wet Basins
SPERT-3	3	Cans	SRS Wet Basins
Sodium Reactor Experiment (Carbide)	1	Can	SRS Wet Basins
CANDU	3	Cans	SRS Wet Basins
CANDU	56	Rods	SRS Wet Basins

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a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

EBR = Experimental Breeder Reactor
HWCTR = Heavy Water Components Test Reactor
CANDU = Canadian Deuterium-Uranium Reactor
LWR = Light Water Reactor
EBWR = Experimental Boiling Water Reactor
ANL-MXOX = Argonne National Laboratory Mixed Oxide
ORNL = Oak Ridge National Laboratory
GCRE = Gas Cooled Reactor Experiment
HTRE = High Temperature Reactor Experiment
ML-1 = Mobile Low Power Plant No. 1
SPERT-3 = Special Power Excursion Test-3
EPRI = Electric Power Research Institute

Table C-7. Radionuclide inventories based on the Reference Fuel Assembly (curies).

Nuclide[a]	Reference Fuel Assembly	Fuel Group					
		A	B	C	D	E	F
H-3	51.6	2540	144,000	46,000	9,090	112	9,780
Kr-85	1,050	51,700	2,920,000	935,000	185,000	2,270	199,000
Sr-89	49.2	2420	137000	43800	8670	107	9320
Sr-90	8,080	398,000	22,500,000	7,200,000	1,420,000	17,500	1,530,000
Y-90	8,080	398,000	22,500,000	7,200,000	1,420,000	17,500	1,530,000
Y-91	213	10,500	593,000	190,000	37,500	461	40,400
Zr-95	454	22,400	1,260,000	404,000	80,000	983	86,000
Nb-95	1,010	49,800	2,810,000	899,000	178,000	2,190	191,000
Nb-95m	3.37	166	9,390	3,000	594	7.30	639
Tc-99	1.03	50.7	2,870	917	181	2.23	195
Rh-103m	1.96	96.6	5,460	1,750	345	4.25	371
Rh-106	21,100	1,040,000	58,800,000	18,800,000	3,720,000	45,700	4,000,000
Ru-103	2.17	107	6,040	1,930	382	4.70	411
Ru-106	21,100	1,040,000	58,800,000	18,800,000	3,720,000	45,700	4,000,000
Ag-110	2.32	114	6,460	2,070	409	5.03	440
Ag-110m	174	8,570	485,000	155,000	30,700	377	33,000
Cd-113m	6.95	342	19,400	6,190	1,220	15.1	1,320
Sn-119m	3.93	194	10,900	3,500	693	8.51	745
Sn-123	14.5	714	40,400	12,900	2,560	31.4	2,750
Sb-125	870	42,900	2,420,000	775,000	153,000	1,880	165,000
Te-125m	212	10,400	590,000	189,000	37,400	459	40,200
Te-127	34.7	1,710	96,600	30,900	6,110	75.2	6,570
Te-127m	35.4	1,740	98,600	31,500	6,240	76.7	6,710
Te-129	0.0012	591	3.34	1.07	0.211	0.0026	0.227
Te-129m	0.00185	911	5.15	1.65	0.326	0.00401	0.351
Cs-134	10,300	507,000	28,700,000	9,170,000	1,810,000	22,300	1,950,000
Cs-137	9,280	457,000	25,800,000	8,260,000	1,640,000	20,100	1,760,000
Ba-137m	8,780	433,000	24,500,000	7,820,000	1,550,000	19,000	1,660,000
Ce-141	0.0646	3.18	180	57.5	11.4	0.140	12.2
Ce-144	47,800	2,350,000	133,000,000	42,600,000	8,420,000	104,000	9,060,000
Pr-144	47,800	2,350,000	133,000,000	42,600,000	8,420,000	104,000	9,060,000
Pr-144m	574	28,300	1,600,000	511,000	101,000	1,240	109,000
Pm-147	18,800	926,000	52,400,000	16,700,000	3,310,000	40,700	3,560,000
Pm-148m	0.00893	0.44	24.9	7.95	1.57	0.0193	1.69
Sm-151	69.4	3,420	193,000	61,800	12,200	150	13,100
Eu-154	727	35,800	2,020,000	647,000	128,000	1,570	138,000
Eu-155	381	18,800	1,060,000	339,000	67,100	825	72,200
Tl-208	8.46	417	23,600	7,530	1,490	18.3	1,600
Pb-209	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Pb-211	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Pb-212	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Bi-211	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Bi-212	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Bi-213	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Po-212	15.1	744	42,100	13,400	2,660	32.7	2,860
Po-213	0.00855	0.421	23.8	7.61	1.51	0.0185	1.62

Table C-7. (continued).

	Reference Fuel	Fuel Group					
Nuclide[a]	Assembly	A	B	C	D	E	F
Po-215	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Po-216	23.6	1,160	65,700	21,000	4,160	51.1	4,470
At-217	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Rn-219	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Rn-220	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Fr-221	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ra-223	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Ra-224	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Ra-225	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ac-225	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ac-227	0.0171	0.842	47.6	15.2	3.01	0.037	3.24
Th-227	0.0164	0.808	45.7	14.6	2.89	0.0355	3.11
Th-228	23.5	1160	65,500	20,900	4,140	50.9	4,450
Th-229	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Th-231	0.0114	0.562	31.8	10.2	2.01	0.0247	2.16
Th-232	0.0172	0.847	47.9	15.3	3.03	0.0373	3.26
Th-234	0.000216	0.0106	0.60	0.192	0.0381	0.000468	0.0409
Pa-231	0.228	11.2	635	203	40.2	0.494	43.2
Pa-233	0.0859	4.23	239	76.5	15.1	0.186	16.3
Pa-234m	0.000216	0.0106	0.60	0.192	0.0381	0.000468	0.0409
U-232	40.9	2,010	114,000	36,400	7,210	88.6	7,750
U-233	39.3	1,940	109,000	35,000	6,930	85.1	7,450
U-234	1.92	94.6	5,350	1,710	338	4.16	364
U-235	0.0114	0.562	31.8	10.2	2.01	0.0247	2.16
U-236	0.0329	1.62	91.6	29.3	5.80	0.0713	6.23
U-237	0.259	12.8	721	231	45.6	0.561	49.1
U-238	0.0842	4.15	235	75.0	14.8	0.182	16.0
Np-237	0.00881	0.434	24.5	7.85	1.55	0.02	1.67
Np-239	9.62	474	26,800	8,570	1,700	21	1,820
Pu-236	112	5,520	312,000	99,700	19,700	250	21,200
Pu-238	51.9	2,560	145,000	46,200	9,150	340	9,830
Pu-239	58	2,860	162,000	51,700	10,200	130	11,000
Pu-240	9,780	482,000	27,200,000	8,710,000	1,720,000	23,000	1,850,000
Pu-241	10,600	522,000	29,500,000	9,440,000	1,870,000	23,000	2,010,000
Am-241	51.7	2,550	144,000	46,000	9,110	450	9,800
Am-242	0.34	16.7	947	303	59.9	0.74	64.4
Am-242m	0.341	16.8	950	304	60.1	0.74	64.6
Am-243	9.62	474	26,800	8,570	1,700	21	1,820
Cm-242	490	24,100	1,360,000	436,000	86,300	1,100	92,800
Cm-243	4.9	241	13,600	4,360	863	11	928
Cm-244	2,750	135,000	7,660,000	2,450,000	485,000	18,000	521,000
Cm-246	0.215	10.6	599	191	37.9	150	40.7
Totals	231,000	11,400,000	644,000,000	206,000,000	40,700,000	520,000	43,800,000

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a. Refer to Table C-8 for the names of the elements.

Table C-8. Chemical symbols used in Table C-7 and the corresponding element names.

H-3 = tritium
Kr = krypton
Sr = strontium
Y = yttrium
Zr = zirconium
Nb = niobium
Tc = technetium
Rh = rhodium
Ru = ruthenium
Ag = silver
Cd = cadmium
Sn = tin
Sb = antimony
Te = tellurium
Cs = cesium
Ba = barium
Ce = cerium
Pr = praseodymium
Pm = promethium
Sm = samarium
Eu = europium
Tl = thallium
Pb = lead
Bi = bismuth
Po = polonium
At = astatine
Rn = radon
Fr = francium
Ra = radium
Ac = actinium
Th = thorium
Pa = protactinium
U = uranium
Np = neptunium
Pu = plutonium
Am = americium
Cm = curium

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APPENDIX D. ACCIDENT ANALYSIS

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APPENDIX D. ACCIDENT ANALYSIS

This appendix provides detailed information on the management of spent nuclear fuel (SNF). The information includes potential accident scenarios for new, modified, and existing facilities that the U.S. Department of Energy (DOE) would use for each alternative. The appendix provides estimates of the quantity and composition of hazardous materials that could be released in an accident as well as the consequences to workers and the public, estimated in terms of dose and latent cancer fatalities for radiological releases and of concentration levels for chemical releases.

The sources of information for the accident analyses for existing facilities are safety analysis reports and basis for interim operation documents. For new or modified facilities the sources vary, depending on the processes involved. In general, DOE performed detailed hazard assessments to identify potential significant accidents, basing the determination of significance on the existence of sufficient energy sources and hazardous materials that, if released, would impact workers or the public. The following sections provide specific information on the hazards assessments for the alternatives.

D.1 General Accident Information

An accident, as discussed in this appendix, is an inadvertent release of radiological or chemical hazardous materials as a result of a sequence of one or more probable events. The sequence usually begins with an initiating event, such as a human error and explosion, or earthquake and structural failure, followed by a succession of other events that could be dependent or independent of the initial event, which dictate the accident's progression and the extent of 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. Examples include equipment or structural failures, human errors, and internal flooding.
- *External initiators* – are independent of facility operations and normally originate from outside the facility. Some external initiators affect the ability of the facility to maintain its confinement of hazardous materials because of potential structural damage. Examples include aircraft crashes, nearby explosions, and toxic chemical releases at nearby facilities that affect worker performance.
- *Natural phenomena initiators* – are natural occurrences that are independent of facility operations and occurrences at nearby facilities or operations. Examples include earthquakes, high winds, floods, lightning, and snow. Although natural phenomena initiators are independent of external facilities, their occurrence can involve those facilities and compound the progression of the accident.

The likelihood of an accident occurring and its consequences usually depend on the initiator and the sequence of events and their frequencies or probabilities. Accidents can be grouped into four categories—anticipated, unlikely, extremely unlikely, and not reasonably foreseeable, as listed in Table D-1.

DOE based the frequencies of accidents at existing SNF management facilities on safety analyses and historical data about event occurrences. For proposed new facilities without design details, DOE based the accident frequencies on hazard analyses, historical data for similar facilities and operations, and best estimates. For all facilities, DOE analyzed the bounding accident in appropriate accident classes (e.g., natural phenomena hazards, operational errors, external events), such as the worst case fire, to represent all other accident in that class.

Table D-1. Accident frequency categories.

Accident Frequency category	Frequency range (occurrences per year)	Description	
Anticipated	Less than once in 10 years but greater than once in 100 years	Accidents that might occur several times during facility lifetime	TC
Unlikely	Less than once in 100 years but greater than once in 10,000 years	Accidents that are not likely to occur during facility lifetime; natural phenomena include Uniform Building Code-level earthquake, maximum wind gust, etc.	
Extremely unlikely	Less than once in 10,000 years but greater than once in 1,000,000 years	Accidents that probably will not occur during facility life cycle; this includes the design-basis accidents	
Beyond extremely unlikely	Less than once in 1,000,000 years	All other accidents	EC TC

Source: DOE (1994).

D.2 Accident Analysis Method

DOE tailored the methods it used to analyze potential accidents to the specific situation. For accidents that could result from operations at existing facilities, the analysis used the applicable impacts described in existing safety analysis documents. For these facilities the analysis included no new modeling; through a screening process, it included only accident scenarios pertaining to operations related to SNF management. Depending on the alternative, one or more new facilities or major modifications to existing facilities could be required. For example, a new Transfer and Storage Facility would be common to many, but not all, of the alternatives. Some alternatives would require the construction of a new treatment component to operate in conjunction with the Transfer and Storage Facility. For these new facilities, hazard analyses were performed to identify bounding accident scenarios, as explained below. The identified accidents were modeled for radiological impacts (Simpkins 1997) using the AXAIRQ computer code (Simpkins 1995a,b), which is described in this section.

The accident sequences analyzed in this EIS would occur at frequencies generally greater than once in 1,000,000 years. However, the analysis considered accident sequences with smaller frequencies if their impacts could provide information important to decisionmaking.

D.2.1 TECHNOLOGIES AND RELATED FACILITIES

DOE has organized the accident data in this appendix by the facilities it would use for each alternative. Table D-2 lists the technologies and the corresponding facilities that DOE would use to implement each. DOE organized the accident impacts in Chapter 4 by technology to reflect potential accident occurrences for the associated facilities listed in Table D-2.

Table D-2 also lists applicable types of fuel that DOE would treat and manage under each alternative. The accident analyses performed for each facility and alternative do not take explicit account of specific fuel properties and characteristics. Rather, the analyses defined a reference fuel assembly (RFA; Appendix C) and furnace batch equivalent (FBE; WSRC 1998) amounts of material at risk (MAR). The FBE MAR was used to analyze all events for all new treatment technologies and the RFA MAR for events related to SRS wet basins and SRS canyons. TC

D.2.2 RADIOLOGICAL HAZARDS

The analysis used the computer code AXAIRQ to model accidental atmospheric radioactive releases from the Savannah River Site (SRS) that are of relatively short duration. AXAIRQ strictly follows the guidance in Regulatory Guide 1.145 (NRC 1982) on accidental releases, and has been verified and validated. The release can originate from a vent or stack and release heights can be from 0 to 100 meters (0 to 328 feet), as appropriate for the scenario. The code considers terrain for elevated releases. In accordance with the regulatory guide, it considers plume meander and fumigation under certain conditions. Plume rise due to buoyancy or momentum is not available. The program uses a 5-year meteorological data base for the Savannah River Site, and determines the shortest distance to the Site boundary in each of the 16 sectors by determining the distance to one of 875 locations along the boundary. The code uses the shortest distance in each sector to calculate the concentration for that sector. DOE used PRIMUS, which was developed by the Oak Ridge National Laboratory, to consider decay and daughter ingrowth.

The analysis assumes that all tritium released would have the form of tritium oxide and, following International Commission on Radiological Protection methodology, the dose conversion factor for tritium has been increased by 50 percent to account for absorption through the skin. For population dose calculations, age-specific breathing rates are applied, but adult dose conversion factors are used. Radiation doses were calculated to the maximally exposed offsite individual (MEI), to the population within 50 miles of the facility, and to an uninvolved worker assumed to be 640 meters (2,100 feet) downwind of the facility.

After DOE calculated the total radiation dose to the public, it used dose-to-risk conversion factors established by the National Council on Radiation Protection and Measurements (NCRP) to estimate the number of latent cancer fatalities that could result from the calculated exposure. No data indicate that small radiation doses cause cancer; however, to be conservative the NCRP assumes that any amount of radiation has some risk of inducing cancer. DOE has adopted the NCRP factors of 0.0005 latent cancer fatality for each person-rem of radiation exposure to the general public and 0.0004 latent cancer fatality for each person-rem of radiation exposure to radiation workers (NCRP 1993).

D.2.3 CHEMICAL HAZARDS

For chemically toxic materials, the long-term health consequences of human exposure to hazardous materials are not as well understood as those related to radiation exposure. A determination of potential health effects from exposures to chemically hazardous materials, compared to radiation, is more subjective. Therefore, the consequences from accidents involving hazardous materials are expressed in terms of airborne concentrations at various distances from the accident location, rather than in terms of specific health effects.

To determine the potential health effects to workers and the public that could result from accidents involving hazardous materials, the airborne concentrations of such materials released during an accident at varying distances from the point of release were compared to the Emergency Response Planning Guideline (ERPG) values (AIHA 1991). The American Industrial Hygiene Association established these values, which depend on the chemical substance, for the following general severity levels to ensure that the necessary emergency actions occur to minimize exposures to humans.

- ERPG-1 Values. Exposure to airborne concentrations greater than ERPG-1 values for a period greater than 1 hour results in an unacceptable likelihood that a person would experience mild transient adverse health effects or perception of a clearly defined objectionable odor.

- ERPG-2 Values. Exposures to airborne concentrations greater than ERPG-2 values for a period greater than 1 hour results in an unacceptable likelihood that a person would experience or develop irreversible or other serious health effects or symptoms that could impair a person's ability to take protective action.

- ERPG-3 Values. Exposure to airborne concentrations greater than ERPG-3 values for a period greater than 1 hour results in an unacceptable likelihood that a person would experience or develop life-threatening health effects.

Not all hazardous materials have ERPG values. For chemicals that do not have ERPG values, a comparison was made to the most restrictive available exposure limits established by other guidelines to control worker accidental exposures to hazardous materials. In this document, the ERPG-2 equivalent that is used is the PEL-TWA (Permissible Exposure Limit – Time Weighted Average) from 29 CFR Part 1910.1000, Subpart Z.

D.3 Impacts of Postulated Accidents Involving Radioactive Materials

TC These sections describe the potential accidents and risks associated with the operation of each facility that may be utilized in the implementation of a technology. The impacts of each technology are shown in Sections D.3.5 to D.3.8. The material at risk in all treatments is the same, only the release fractions change. For these cases, over 95 percent of the calculated doses come from the release of plutonium-240 and curium-244. The only exception to this are criticality accident scenarios when over 99 percent of the dose comes from the release of ruthenium-106.

D.3.1 H-CANYON AND FB-LINE

Tables D-3 and D-4 summarize potential accidents and their impacts for the H-Canyon and FB-Line facilities, respectively (WSRC 1993, 1995; TtNUS 1999b). TC

D.3.2 RECEIVING BASIN FOR OFFSITE FUEL

Potential accidents and their impacts for the Receiving Basin for Offsite Fuel (RBOF) facility have been documented in a safety analysis report (WSRC 1997a). Table D-5 lists the accidents with the highest risks and consequences.

D.3.3 REACTOR DISASSEMBLY BASIN

Potential accidents and their impacts for the L-Reactor Disassembly Basin have been documented in a basis for interim operation report (WSRC 1997b). Table D-6 summarizes the results.

D.3.4 TRANSFER AND STORAGE FACILITIES

DOE could collocate the transfer and storage facilities either in separate buildings or in a single building. The accident impacts associated with the operation of these facilities apply to both cases and assume the location of the facilities in L-Area.

D.3.4.1 Transfer Facility Accidents

Radioactive Material Leaks From Shipping or Storage Cask (TS-1)

Radioactive materials could leak from shipping or storage casks. In this accident sequence, radioactive material would leak to the surface of the shipping or storage cask and a small amount would become airborne. The principal radionuclides would be cesium-137, cerium-144, ruthenium-106, and strontium-90. The total curies released in this scenario would 1.0×10^{-7} and would result in negligible consequences. This event is postulated to occur once in 10 years of operation. The calculated consequences for this scenario are listed in Table D-7.

Table D-3. H-Canyon radiological accidents and impacts.[a]

	Accident consequences

TC

Accident	Maximum curies released	Accident frequency	Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Ruthenium volatilization	140	Once in 11 years	0.13	0.013	770	0.39
Fire	0.57	Once in 1,600 years	0.53	0.055	3,300	1.6
Earthquake	860	Once in 5,000 years	1.8	0.246	14,000	7.0
Coil and tube, cooling tower circulated	13	Once in 14,000 years	13	1.3	78,000	39
Transfer error to Building 211-H	3,700	Once in 14,000 years	1.5	0.16	9,200	4.6
Hydrogen deflagration	1.1	Once in 18,000 years	1.0	0.11	6,400	3.2
Criticality	47,000	Once in 77,000 years	0.029	0.0012	18	0.009

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a. Source: TtNUS (1999b).
b. MEI = Maximally exposed individual.

Table D-4. FB-Line radiological accidents and impacts.

			Accident Consequences			
Accident	Maximum curies released	Accident frequency	Noninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Design basis earthquake, 0.2g intensity[b]	0.31	Once in 5,000 years	0.34	0.042	150	0.077
Propagated fire[c]	2.2	Once in 59,000 years	0.18	0.14	1,100	0.53

---

a. MEI = Maximally exposed individual.
b. Source: WSRC (1993).
c. Source: WSRC (1995).

Cask Decontamination Waste Released to Environment (TS-3)

Casks would be washed at receipt and before shipping. The wash liquid probably would be collected in a sump or storage tank and released to the environment if sample results showed contamination levels to be within acceptable limits. Excessively radioactive or hazardous material could be pumped inadvertently to an outfall rather than to the liquid radioactive waste system or hazardous waste storage if there was an error in processing samples or reading laboratory test results. This scenario assumes that happens and a small amount becomes airborne. The total curies released to air would be 2.0×10^{-7} and would result in negligible consequences. This event is postulated to occur once in 100 years of operation. The calculated consequences for this scenario are listed in Table D-7.

EC

Table D-5. Receiving Basin for Offsite Fuel radiological accidents and impacts.[a]

			Accident Consequences			
Accident	Maximum curies released	Accident frequency	Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Fuel rupture	160,000	Once in 1.4 years	0.0018	1.9×10^{-4}	10	0.005
Resin explosion	1.0	Once in 400 years	0.0012	1.3×10^{-4}	7.8	0.0039
Uncontrolled chemical reaction	1,600,000	Once in 450 years	0.018	0.0019	100	0.05
Resin fire	11	Once in 1,200 years	1.3×10^{-4}	1.4×10^{-5}	0.83	4.2×10^{-4}
Process-induced criticality	4,800	Once in 1,500 years	0.16	0.016	970	0.49
NPH[d] (high winds)						
Fuel breach	1,600,000	Once in 2,600 years	0.13	0.0024	130	0.063
Criticality	48,000	Once in 26,000 years	13	0.22	12,000	6.2
NPH (earthquake)						
Waste tank breach	0.69	Once in 280 years	0.0065	1.1×10^{-4}	6.3	3.2×10^{-3}
Fuel breach	1,600,000	Once in 36,000,000 years	0.13	0.0024	130	0.063
Criticality	48,000	Once in 38,000,000 years	13	0.22	12,000	6.2

TC

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a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.
c. Data not available.
d. NPH = Natural Phenomenon Hazard.

Table D-6. Reactor Disassembly Basins radiological accidents and impacts.[a]

			Accident Consequences			
Accident	Maximum curies released	Accident frequency	Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Basin overfill	(c)	Once in 100 years	0	4.6×10^{-4}	(c)	
Mishandling fuel assembly	(c)	Once in 100 years	25	0	0	0
Criticality	4,800	Once in 300 years	0.16	0.016	660	0.3
Basin water draindown	(c)	Once in 500 years	0.055	0.016	(c)	

TC

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a. Source: WSRC (1997b), TtNUS (1999a). TC
b. MEI = Maximally exposed individual.
c. Accidents expected to result in low consequences and risks to the onsite worker population and the offsite population. Quantitative estimates of consequences and risks are not available.

Table D-7. Transfer Facility radiological accidents and impacts.[a]

			Accident Consequences			
Accident	Maximum curies released	Accident frequency	Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Railroad car contamination, TS-1	1.0×10^{-7}	Once in 10 years	1.7×10^{-7}	1.8×10^{-8}	7.4×10^{-4}	3.7×10^{-7}
Sump Release, TS-3	2.1×10^{-7}	Once in 100 years	3.5×10^{-7}	3.6×10^{-8}	1.5×10^{-3}	7.4×10^{-7}

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a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.

Cask Criticality From Internal Disruption - Single Shipping Cask (TS-4)

Shipping casks containing spent fuel would be moved between facilities by rail or truck and loaded or unloaded from transports using overhead or mobile cranes. The casks contain internal structures that maintain fuel separation or provide neutron absorption. Casks certified in accordance with regulations of the U.S. Department of Transportation are designed to withstand drops from a specified height. However, if the cask internal structures were not assembled properly, disruption and redistribution of the fuel could occur. In addition, the fuel itself could be damaged or loose its integrity and redistribute itself in a critical configuration. This could produce a criticality at the time of the disruption or later if the cask were filled with water for purging. If this event occurred outdoors, the release would not be filtered. This event is not credible.

Criticality From Fuel Dropped On To Floor Or In To Dry Storage Rack (TS-5)

A criticality accident could result if spent nuclear fuel were dropped in a pile on the floor or dropped into the cask-unloading dry storage rack. The fuel drop could result from operator error or equipment failure in the handling ma-chine or spent fuel structure. Double contingency protection would require the dropping of at least two spent fuel loads (assemblies, canisters, bundles, etc.) before a criticality occurred. In addition, the first drop would have to be unrecovered when the second drop occurred. Procedures would require the recovery of the first fuel dropped before material movement could resume. The effects of a criticality event would be mitigated by shielding and the physical distance of the operators in remote handling operations. This event would occur inside the facility, so released radionuclides would be filtered. This event is not considered credible.

Criticality of Spent Fuel in Several Adjacent Shipping Casks (TS-7)

This event addresses a criticality accident among spent nuclear fuel brought together in multiple shipping or storage casks. Because the nuclear reaction would occur in all the fuel in the array, several casks could be involved. A criticality accident of this nature would produce a direct radiation hazard and would release radioactive contamination if one or more casks were breached. However, a criticality during cask dry storage is not a credible event.

D.3.4.2 Dry Storage Facility Accidents

Spent Nuclear Fuel Dry Storage Process-Related Criticality Accident (SLS-2)

This accident scenario involves criticality resulting from the improper loading of dry storage racks or the mechanical failure of racks. Mechanical failure or collapse could result from a crane impact or structural flaw in the racks. Improper loading would result in sufficient spent nuclear fuel assemblies placed near one another with insufficient neutron absorbers to prevent a criticality. A collapse of the racks would result in sufficient spent fuel assemblies piled near one another in the debris with insufficient neutron absorbers to prevent a criticality. This event is TC postulated to occur once in 330 years of operation. The calculated consequences for this scenario are listed in Table D-8.

Natural Phenomenon Hazard-Induced Spent Nuclear Fuel Dry Storage Criticality Accident (SLS-3)

This accident scenario involves a natural phenomenon hazard-induced criticality resulting from an earthquake-induced mechanical failure of racks (e.g., collapse or crane impact and collapse) or a subsequent fission product release resulting from a fuel breach. This event is predicated on the assumption that the facility could withstand an earthquake intact and operational. However, the scenario assumes that the structure that contains the material at risk fails, resulting in the event. This event is postulated to occur once TC in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-8.

Spent Nuclear Fuel Dry Storage Fission Product Release (SLS-1)

This accident scenario involves the release of fission products from a fuel breach and a simultaneous loss of confinement due to an earthquake. The fuel breach would result from an earthquake-induced mechanical failure or collapse of the storage racks or an earthquake-induced crane impact. This event is postulated to TC occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-8.

D.3.5 ELECTROMETALLURGICAL TREATMENT ACCIDENT SEQUENCES

Melter Eruption (GFP-1 and MM-1)

The electrometallurgical technology would have two separate melters. The melter eruption postulated event could result from impurities in the glass melt (GFP-1) or the metal melt (MM-1). Impurities could range from water that could cause a steam eruption to chemical contaminants that could react at elevated temperatures and produce a highly exothermic reaction (eruption or deflagration). The scenario assumes that the resulting sudden pressure increase would eject the fissile-material-bearing melt liquid into the processing structure. It also assumes that the energy release would not damage the processing structure and its associated filtered exhaust ventilation system. The melter ventilation systems would remove or dilute explosive mixtures that could build up in the gas space above the molten material. Operating procedures and verifications would prevent the addition of impure or incorrect materials to the melt. Therefore, this event is postulated to occur once in 20 years of operation. TC

If the eruption was large, operating personnel would hear and see it. A small eruption might be detected only by airborne radiation monitors because the remotely-operated melters would be in a heavily shielded area. The effects of the eruption would be mitigated by the melter design, which would include venting methods to respond to an over-pressure event. The melter building structure and the ventilation system would confine particulate radioactive material released in the eruption. The calculated consequences for this scenario are listed in Table D-9 for the molten glass release and Table D-10 for the molten metal spill. Noble gases and tritium released on the event would not be filtered. EC

Earthquake-Induced Fission Product Release and Confinement Failure (GFP-4 and MM-5)

The fission-product-release scenario involves damage to the melter structure and its associated systems that would release fission products. This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 2,000 years of operation. The calculated

Table D-8. Dry Storage radiological accidents and impacts.[a]

Accident	Maximum curies released	Accident frequency	Accident Consequences: Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Criticality in storage, SLS-2	4,800	Once in 330 years	0.16	0.016	660	0.3
Earthquake-induced criticality, SLS-3	48,000	Once in 2,000 years	13	0.22	12,000	6.2
Fuel breach during earthquake, SLS-1	1,100,000	Once in 2,000 years	0.014	0.0015	54.1	0.027

TC

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a. Source: TtNUS (1999a).
b. MEI = Maximally expose individual.

Table D-9. Electrometallurgical Treatment radiological accidents and impacts (glass melter only).[a]

Accident	Maximum curies released	Accident frequency	Accident Consequences: Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Melter eruption, GFP-1	1,200	Once in 20 years	1.6×10^{-5}	1.1×10^{-6}	0.04	2.0×10^{-5}
Earthquake-induced fission product release and confinement failure, GFP-4	2,300	Once in 2,000 years	3.2×10^{-5}	2.2×10^{-6}	0.08	4.0×10^{-5}
Melter eruption with loss of ventilation, GFP-1a	1,200	Once in 2,000 years	0.002	2.3×10^{-4}	9.5	0.0047
Earthquake spill with loss of ventilation, GFP-4a	2,300	Once in 200,000 years	0.038	6.2×10^{-4}	26	0.013

TC

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a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.

Table D-10. Melt and Dilute Treatment radiological accidents and impacts (these accidents also apply to the metal melter for Electrometallurgical Treatment).[a]

Accident	Maximum curies released	Accident frequency	Accident Consequences: Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Melter eruption, MM-1	0.09	Once in 20 years	7.1×10^{-6}	7.4×10^{-7}	0.03	1.5×10^{-5}
Criticality due to multiple batching 5×10^{17} fissions, MM-4	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release and confinement failure, MM-5	2,300	Once in 2,000 years	6.8×10^{-5}	5.9×10^{-6}	0.23	1.2×10^{-4}
Melter eruption with loss of ventilation, MM-1a	9,200	Once in 2,000 years	0.71	0.074	3,000	1.5
Process criticality with loss of ventilation, MM-4a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced spill with loss of ventilation, MM-5a	21,000	Once in 200,000 years	30	0.50	21,000	10

TC

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a. Source: TtNUS (1999a), TtNUS (2000).

b. MEI = Maximally exposed individual.

consequences for this scenario are listed in Table D-9 for the glass melt and Table D-10 for the metal melt.

Earthquake-Induced Fission Product Release and Confinement Failure (GFP-4 and MM-5)

The fission-product-release scenario involves damage to the melter structure and its associated systems that would release fission products. This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-9 for the glass melt and Table D-10 for the metal melt.

Criticality Accident (MM-4)

Melter design volume limits would prevent a criticality accident. However, to preserve flexibility of operation, there would have to be provision for some excess volume, so a criticality accident could result from charging multiple batches of fissile material to the metal melter. The batching operation would be a procedure-guided operator action. "Double Contingency" would require a second operator to verify the processing steps independently. Therefore, this event is postulated to occur once in 330 years of operation. In the event of a criticality, the process building structure and filtered exhaust system would remain intact and would confine fission products and shield against direct radiation exposure. The calculated consequences for this scenario are listed in Table D-10. TC

Melter Eruption with Coincident Ventilation Failure (GFP-1a and MM-1a)

TC

This scenario has the same initiating event as the glass melt eruption (GFP-1) or the metal melt eruption (MM-1) but with a coincident failure of the HEPA filtration system. As this event requires both a melter eruption and a ventilation failure, the postulated frequency for this event is once in 2,000 years. The calculated consequences for this scenario are listed in Table D-9 for the molten glass release and Table D-10 for the molten metal spill.

Earthquake-Induced Fission Product Release and Confinement Failure with Coincident Ventilation Failure (GFP-4a and MM-5a)

This scenario has the same initiating event as the glass melt spill (GFP-4) or the metal melt spill (MM-5) but with a coincident failure of the HEPA filtration system. As this event requires both a seismic event and a ventilation failure, the postulated frequency for this event is once in 200,000 years. The calculated consequences for this scenario are listed in Table D-9 for the molten glass release and Table D-10 for the molten metal spill. TC

Criticality Accident with Coincident Ventilation Failure (MM-4a)

This scenario has the same initiating event as the criticality accident (MM-4), but with a coincident failure of the HEPA filtration system. As this event requires both a criticality and a ventilation failure, the postulated frequency for this event is once in 33,000 years. The calculated consequences for this scenario are listed in Table D-10.

<u>Other Accident Scenarios</u>

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.6 MELT AND DILUTE TREATMENT FACILITY ACCIDENT SEQUENCES

The accidents for the Melt and Dilute Treatment Facility would be the same for either a new facility or in a renovated reactor building. TC

Melter Eruption (MM-1)

This event would be identical to the metal melt eruption described as MM-1 in D.3.5. The calculated consequences are presented in Table D-10.

Criticality Accident (MM-4)

The criticality event would be identical to that described for the metal melter as MM-4 in D.3.5. The calculated consequences are presented in Table D-10.

Earthquake-Induced Fission Product Release and Confinement Failure (MM-5)

The fission product release and confinement failure would be identical to that described as MM-5 in D.3.5 The calculated consequences are presented in Table D-10.

Melter Eruption with Ventilation Failure (MM-1a)

This event would be identical to the metal melt eruption with ventilation failure described in D.3.5. The calculated consequences are presented in Table D-10.

Earthquake-Induced Fission Product Release and Confinement Failure with Coincident Ventilation Failure (MM-5a)

This event would be identical to the Earthquake-induced spill with ventilation failure described in D.3.5. The calculated consequences are presented in Table D-10.

Criticality Accident with Coincident Ventilation Failure (MM-4a)

This event would be identical to the Double Batching Criticality with ventilation failure described in D.3.5. The calculated consequences are presented in Table D-10.

<u>Other Accident Scenarios</u>

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.7 MECHANICAL DILUTION TREATMENT

D.3.7.1 <u>Press and Dilute Treatment Accident Sequences</u>

Fission Product Release (SDP-2)

This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-11.

Spent Nuclear Fuel-Depleted Uranium Press Process Criticality Accident (SDP-3)

This process-related criticality would result from multiple batches of spent fuel plates introduced into the press or an inadvertent substitution of spent fuel plates for depleted uranium plates. In either instance sufficient spent fuel in the configuration would result in a criticality. This event is postulated to occur once in 330 years of operation. The calculated consequences for this scenario are listed in Table D-11.

Earthquake-Induced Fire/Pyrophoric Reaction (SDP-4)

An earthquake-induced fire or pyrophoric reaction would result from friction due to mechanical shredding, electrical or mechanically induced fires on uranium metal fuel, or a fire started by a hydraulic fluid leak that resulted in a subsequent pyrophoric reaction. This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 20,000 years of operation. The calculated consequences for this scenario are listed in Table D-11.

Table D-11. Press and Dilute radiological accidents and impacts.[a]

			Accident Consequences			
Accident	Maximum curies released	Accident frequency	Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
SNF-DU press process criticality, SDP-3	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake induce fission product release, SDP-2	230	Once in 2,000 years	3.2×10^{-6}	2.2×10^{-7}	8.0×10^{-3}	4.0×10^{-6}
Earthquake-induced fire/pyrophoric reaction, SDP-4	2,300	Once in 20,000 years	3.6×10^{-5}	2.6×10^{-6}	0.095	4.8×10^{-5}
Process criticality with loss of ventilation, SDP-3a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced fission product release with loss of ventilation, SDP-2a	240	Once in 200,000 years	0.010	1.6×10^{-4}	6.6	0.0033

TC

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a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.

Fission Product Release with Coincident Ventilation Failure (SDP-2a)

This scenario has the same initiating event as the Fission Product Release Accident (SDP-2), but with a coincident failure of the HEPA filtration system. As this event requires both an earthquake and a ventilation failure, the postulated frequency for this event is once in 200,000 years. The calculated consequences for this scenario are listed in Table D-11.

TC

Spent Nuclear Fuel-Depleted Uranium Press Process Criticality Accident (SDP-3a)

This scenario has the same initiating event as the criticality accident (SDP-3), but with a coincident failure of the HEPA filtration system. As this event requires both a criticality and a ventilation failure, the postulated frequency for this event is once in 33,000 years. The calculated consequences for this scenario are listed in Table D-11.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.7.2 Chop and Dilute Treatment Accident Sequences

Fission Product Release (SS-2)

This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-12.

TC

Spent Nuclear Fuel Shredding Process Criticality Accident (SS-3)

This process-related criticality would result from feeding multiple batches of spent fuel to the fuel

TC

shredder. This would result in sufficient spent fuel in a configuration that could result in a criticality. This event is postulated to occur once in 330 years of operation. The calculated consequences for this scenario are listed in Table D-12.

Earthquake-Induced Fire/Pyrophoric Reaction (SS-4)

An earthquake-induced fire or pyrophoric reaction could result from friction due to mechanical shredding, electrical or mechanically induced fires on uranium metal fuel, or a fire started by a hydraulic fluid leak that resulted in a subsequent pyrophoric reaction. This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would fail, resulting in the event. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-12.

Table D-12. Chop and Dilute radiological accidents and impacts.[a]

TC

Accident	Maximum curies released	Accident frequency	Accident Consequences: Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Process criticality, SS-3	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release, SS-2	0.07	Once in 2,000 years	1.2×10^{-7}	1.2×10^{-8}	4.9×10^{-4}	2.5×10^{-7}
Earthquake-induced fire, SS-4	2.3	Once in 2,000 years	3.6×10^{-6}	3.8×10^{-7}	0.015	7.7×10^{-6}
Process criticality with loss of ventilation, SS-3a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake release with loss of ventilation, SS-2a	66	Once in 200,000 years	0.012	0.0012	49	0.024
Earthquake-induced fire with loss of ventilation, SS-4a	2,100	Once in 200,000 years	3	0.050	2,100	1.0

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a. Source: TtNUS (1999a).

b. MEI = Maximally exposed individual.

Fission Product Release with Coincident Ventilation Failure (SS-2a)

This scenario has the same initiating event as the Fission Product Release Accident (SS-2), but with a coincident failure of the HEPA filtration system. As this event requires both an earthquake and a ventilation failure, the postulated frequency for this event is once in 200,000 years. The calculated consequences for this scenario are listed in Table D-12.

TC

TC

Spent Nuclear Fuel Shredding Process Criticality Accident (SS-3a)

This scenario has the same initiating event as the criticality accident (SS-3), but with a coincident failure of the HEPA filtration system.

As this event requires both a criticality and a ventilation failure, the postulated frequency for this event is once in 33,000 years. The calculated consequences for this scenario are listed in Table D-12.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.8 VITRIFICATION FACILITIES

D.3.8.1 Glass Material Oxidation and Dissolution System Accident Scenarios

Melter Eruption (GMF-1)

The postulated melter eruption could result from impurities in the metal melt. Impurities could range from water that could cause a steam eruption to chemical contaminants that could react at elevated temperatures and produce a highly exothermic reaction (eruption or deflagration). This scenario assumes that the resulting sudden pressure increase would eject the fissile-material-bearing melt liquid into the processing structure. It also assumes that the energy release would not damage the processing structure and its associated filtered exhaust ventilation system. The melter offgas and inerting systems would remove or dilute explosive mixtures that might build up in the gas space above the molten material. Operating procedures and verifications prevent the addition of impure or incorrect materials to the melt. Therefore, this event is postulated to occur once in 20 years of operation.

If a large eruption did occur, the appearance and sound would alert operating personnel. A small eruption might be detected only by airborne radiation monitors because the remotely operated melters would be in a heavily shielded area. The effects of the eruption would be mitigated by the design of the melter, which would include venting methods to respond to an over-pressure event. The melter building structure and the ventilation system would confine particulate radioactive material released in the eruption. The calculated consequences for this scenario are listed in Table D-13.

Criticality Accident (GMF-4)

Melter design volume limits would prevent a criticality accident. However, to preserve flexibility of operation, there would have to be provision for some excess volume, a criticality accident could result from charging multiple batches of fissile material to the metal melter. The batching operation would be a procedure-guided operator action. “Double Contingency” would require a second operator to verify the processing steps independently. Therefore, this event is postulated to occur once in 33,000 years of operation. In the event of a criticality, the process building structure and filtered exhaust system would remain intact and would confine fission products and shield against direct radiation exposure. The calculated consequences for this scenario are listed in Table D-13.

Earthquake-Induced Fission Product Release and Confinement Failure (GMF-5)

The fission-product-release scenario involves damage to the melter structure and its associated systems that would release fission products. This event assumes that the facility would withstand an earthquake and remain operational. However, it also assumes that the structure that contains the material at risk would to fail, resulting in the release. This event is postulated to occur once in 2,000 years of operation. The calculated consequences for this scenario are listed in Table D-13.

Table D-13. GMODS radiological accidents and impacts.[a]

TC

			Accident Consequences			
Accident	Maximum curies released	Accident frequency	Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Melter eruption, GMF-1	1,200	Once in 20 years	1.6×10^{-5}	1.1×10^{-6}	0.04	2.0×10^{-5}
Criticality due to multiple batching 5×10^{17} fissions, GMF-4	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release and confinement failure, GMF-5	2,300	Once in 2,000 years	3.3×10^{-5}	2.2×10^{-6}	0.08	4.0×10^{-5}
Melter eruption with loss of ventilation, GMF-1a	1,200	Once in 2,000 years	0.0024	2.6×10^{-4}	10	0.0052
Process criticality with loss of ventilation, GMF-4a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced release with loss of ventilation, GMF-5a	2,300	Once in 200,000 years	0.041	6.8×10^{-4}	28	0.014

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a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.

Melter Eruption with Coincident Ventilation Failure (GMF-1a)

This scenario has the same initiating event as the melter eruption (GMF-1), but with a coincident failure of the HEPA filtration system. As this event requires both a melter eruption and a ventilation failure, the postulated frequency for this event is once in 2,000 years. The calculated consequences for this scenario are listed in Table D-13.

Earthquake-Induced Fission Product Release and Confinement Failure with Coincident Ventilation Failure (GMF-4a)

This scenario has the same initiating event as the earthquake-induced melt spill (GMF-4), but with a coincident failure of the HEPA filtration system. As this event requires both a seismic event and a ventilation failure, the postulated frequency for this event is once in 200,000 years. The calculated consequences for this scenario are listed in Table D-13.

Criticality Accident with Coincident Ventilation Failure (GMF-4a)

This scenario has the same initiating event as the criticality accident (GMF-4), but with a coincident failure of the HEPA filtration system. As this event requires both a criticality and a ventilation failure, the postulated frequency for this event is once in 33,000 years. The calculated consequences for this scenario are listed in Table D-13.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.8.2 Plasma Arc Accident Scenarios

Melter Eruption (GMF-1), Criticality Accident (GMF-4), and Earthquake-Induced Fission Product Release and Confinement Failure (GMF-5) and Corresponding Events with Coincident Loss of Ventilation (GMF-1a, 4a, and 5a)

TC

These events are identical in description to GMF-1, GMF-1a, GMF-4, GMF-4a, GMF-5, and GMF-5a as described in D.3.8.1 for the Glass Material Oxidation and Dissolution System accident scenarios. The calculated consequences for these scenarios are listed in Table D-14.

TC

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.8.3 F Canyon Dissolve and Vitrify Treatment Accident Sequences

Melter Eruption (GMF-1), Criticality Accident (GMF-4), and Earthquake-Induced Fission Product Release and Confinement Failure (GMF-5) and Corresponding Events with Coincident Loss of Ventilation (GMF-1a, 4a, and 5a)

TC

These events are identical in description to GMF-1, GMF-1a, GMF-4, GMF-4a, GMF-5, and GMF-5a as described in D.3.8.1 for the Glass Material Oxidation and Dissolution System accident scenarios. The calculated consequences for these scenarios are listed in Table D-15.

TC

Table D-14. Plasma Arc radiological accidents and impacts.[a]

TC

			Accident Consequences			
Accident	Maximum curies released	Accident frequency	Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Melter eruption, GMF-1	1,200	Once in 20 years	1.6×10^{-5}	1.1×10^{-6}	0.040	2.0×10^{-5}
Criticality due to multiple batching 5×10^{17} fissions, GMF-4	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release and confinement failure, GMF-5	2,300	Once in 2,000 years	3.3×10^{-5}	2.2×10^{-6}	0.080	4.0×10^{-5}
Melter eruption with loss of ventilation, GMF-1a	1,200	Once in 2,000 years	0.0062	6.4×10^{-4}	26	0.013
Process criticality with loss of ventilation, GMF-4a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced release with loss of ventilation, GMF-5a	2,400	Once in 200,000 years	0.10	0.0017	71	0.035

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a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.

Table D-15. F Canyon Dissolve and Vitrify radiological accidents and impacts.[a]

TC

			Accident Consequences			
Accident	Maximum curies released	Accident frequency	Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Glass melt eruption, GMF-1	1,200	Once in 20 years	1.3×10^{-5}	1.2×10^{-6}	0.044	2.2×10^{-5}
Criticality due to multiple batching, GMF-4	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release and confinement failure, GMF-5	2,300	Once in 2,000 years	2.5×10^{-5}	2.4×10^{-6}	0.088	4.4×10^{-5}
Glass melt eruption with loss of ventilation, GMF-1a	1,200	Once in 2,000 years	0.0019	2.8×10^{-4}	11	0.0056
Process criticality with loss of ventilation, GMF-4a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced release with loss of ventilation, GMF-5a	2,300	Once in 200,000 years	0.051	8.1×10^{-4}	32	0.016

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a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.3.8.4 New Dissolve and Vitrify Facility

TC

Melter Eruption (GMF-1), Criticality Accident (GMF-4) and Earthquake-Induced Fission Product Release and Confinement Failure (GMF-5) and Corresponding Events with Coincident Loss of Ventilation (GMF-1a, 4a, and 5a)

TC

These events are identical in description to GMF-1, GMF-1a, GMF-4, GMF-4a, GMF-5, and GMF-5a as described in D.3.8.1 for the Glass Material Oxidation and Dissolution System accident scenarios. The calculated consequences for these scenarios are listed in Table D-16.

Other Accident Scenarios

Other accident scenarios were considered. However, these accident sequences are not listed here as they had the same or lower consequences as listed accidents, though with a much lower accident frequency.

D.4 Comparison of Accident Impacts for Alternative Facility Locations

An alternative location for new facilities would be C-Area. If DOE located the facilities in C-Area, doses to the MEI and the population would be expected to be approximately 4.0 percent and 12 percent higher, respectively, due to the shorter distance to the Site boundary.

D.5 Impacts of Postulated Accidents Involving Nonradioactive Hazardous Materials

This section summarizes the impacts of potential accidents involving hazardous chemicals at the Receiving Basin for Offsite Fuel as documented in the safety analysis report for the facility (WSRC 1995). These accidents would not involve radioactive materials.

Table D-16. New Dissolve and Vitrify radiological accidents and impacts.[a]

TC

			Accident Consequences			
Accident	Maximum curies released	Accident frequency	Noninvolved worker (rem)	MEI[b] (rem)	Offsite population (person-rem)	Latent cancer fatalities
Glass melt eruption, GMF-1	1,200	Once in 20 years	1.6×10^{-5}	1.1×10^{-6}	0.04	2.0×10^{-5}
Criticality due to multiple batching GMF-4	4,700	Once in 330 years	0.004	4.8×10^{-5}	1.6	0.0008
Earthquake-induced fission product release and confinement failure, GMF-5	2,300	Once in 2,000 years	3.3×10^{-5}	2.2×10^{-6}	0.08	4.0×10^{-5}
Melter eruption with loss of ventilation, GMF-1a	1,200	Once in 2,000 years	0.0024	2.6×10^{-4}	10	0.0052
Process criticality with loss of ventilation, GMF-4a	14,000	Once in 33,000 years	0.71	0.074	3,000	1.5
Earthquake-induced release with loss of ventilation GMF-5a	2,300	Once in 200,000 years	0.041	6.8×10^{-4}	28	0.014

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a. Source: TtNUS (1999a).
b. MEI = Maximally exposed individual.

The hazard analysis documented in the safety analysis report identified three chemical spill events that required unique accident analyses. This section describes the analysis of these events, which include chemical spills of sodium hydroxide, nitric acid, and sodium nitrite from storage dumpsters outside the facility.

D.5.1 LOSS OF 50-PERCENT SODIUM HYDROXIDE CONTAINMENT

Sodium hydroxide (NaOH), used for anion resin regeneration, is stored in a skid-mounted 1,000-gallon dumpster on a chemical pad west of Building 245-H. This dumpster is typically filled to 900 gallons and is heated during the winter to approximately 10 to 12°F above the crystallization point using 25-psi steam routed through piping inside the dumpster.

If an initiating event occurred that ruptured the tank, the chemical would accumulate in the bermed area of the pad. The rate of leakage from the dumpster would depend on the point of the breach and the severity of the opening. A worst-case breach would drain the contents of the dumpster within minutes. This scenario takes no credit for the berm containing the chemical spill. Therefore, the sodium hydroxide would spread over a large area, which would result in a larger airborne release rate than would a bermed release.

The sodium hydroxide plume would be transported by the wind as tiny particles. Therefore, a Gaussian plume model is appropriate. This event is postulated to occur once in 190 years of operation.

The calculated concentration would be lower than the lowest concentration guideline (PEL-TWA) for either on- or offsite. Therefore, the consequences of the release would be insignificant and there is no need for further analysis at greater distances.

D.5.2 LOSS OF 50-PERCENT NITRIC ACID CONTAINMENT

DOE uses nitric acid (HNO_3) in the regeneration of cation resin, and stores it in a skid-mounted 1,000-gallon dumpster west of Building 245-H. The dumpster is typically filled to 900 gallons. Nitric acid is supplied to the Resin Regeneration Facility through underground piping. The chemical pad is approximately at ground level outside the Receiving Basin for Offsite Fuel and the Resin Regeneration Facility. It is surrounded by a dike to contain spills.

If an initiating event occurred that ruptured the tank, the chemical would accumulate in the bermed area of the pad. The rate of leakage from the dumpster would depend on the point of the breach and the severity of the opening. A worst-case breach would drain the contents of the dumpster within minutes. This scenario takes no credit for the berm containing the chemical spill. Therefore, the nitric acid would spread over a large area, which would result in a larger airborne release rate than would a bermed release. This event is postulated to occur once in 190 years of operation.

The release would result in a concentration of 3.1×10^{-3} milligrams per cubic meter at 640 meters (2,100 feet) and 4.0×10^{-4} milligrams per cubic meter at the nearest Site boundary. These values are both lower than the Emergency Response Planning Guideline-2 values.

D.5.3 LOSS OF 30-PERCENT SODIUM NITRITE CONTAINMENT

DOE stores sodium nitrite ($NaNO_2$), a waste tank corrosion inhibitor, in a skid-mounted 1,000-gallon dumpster on a chemical pad west of Building 245-H. The contents of the dumpster are pumped to an adjacent Holdup Tank with a maximum capacity of 1,600 gallons. This analysis assumes that the contents of both tanks are filled to their total combined volume of 2,600 gallons. The chemical pad is approximately at ground level outside the Receiving Basin for Offsite Fuel and the Resin Regeneration Facility, and is surrounded by a dike.

If an initiating event occurred that ruptured the tank, the chemical would accumulate in the bermed area of the pad. The rate of leakage from the dumpster or holdup tank would depend on the point of the breach and the severity of the opening. A worst-case breach would drain the contents of the dumpster or holdup tank within minutes. This scenario takes no credit for the berm containing the chemical spill. Therefore, the sodium nitrite would spread over a large area, which would result in a larger airborne release rate than would a bermed release. This event is postulated to occur once in 180 years of operation.

The calculated airborne concentration at a downwind distance of 100 meters (328 feet) would be 0.006 milligrams per cubic meter, which is less than the lowest concentration guideline (PEL-TWA). Therefore, the consequences of the release would be insignificant, and there is no need for analysis at greater distances.

D.5.4 SURFACE VEHICLE IMPACT

The impact of a surface vehicle with a chemical dumpster has been identified as a potential initiating event for chemical leakage. The consequences of the events would be the same as the consequences for the events analyzed in Sections D.5.1 through D.5.3. The postulated frequency for each of these chemical releases from surface vehicle impact would be once in 3,400 years.

D.6 Environmental Justice

In the event of an accidental release of radioactive or hazardous chemical substances, the dispersion of such substances would depend on meteorology conditions, such as wind direction, at the time. Given the variability of meteorology conditions and the low probability and risk of accidents, an accident would be unlikely to occur that would result in disproportionately high or adverse human health and environmental impacts to minorities or low-income populations.

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Table D-2. Alternatives and corresponding facilities.

Facility	Direct co-disposal	Repackage and prepare to ship	Melt and dilute	Melt and dilute in a renovated reactor	Mechanical dilution	Vitrification technologies	Electrometal-lurgical treatment	Conventional processing	Continued wet storage
F and H Canyons[a]						Yes		Yes	
Receiving Basin for Offsite Fuel[a]	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Reactor Disassembly Basin[a]	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Transfer and Storage Facility	Yes	Yes	Yes	Yes	Yes	Yes	Yes		
Electrometallurgical Treatment Facility							Yes		
Melt and Dilute Treatment Facility			Yes						
Melt and Dilute in a Renovated Reactor[a]				Yes					
Mechanical Dilution Treatment Facility					Yes[b]				
Vitrification Facilities:									
Glass Material Oxidation and Dissolution System						Yes[c]			
Plasma Arc						Yes[c]			
Dissolve and Vitrify, New Facility						Yes[d]			
Dissolve and Vitrify, F Canyon						Yes[d]			
Applicable Fuel Type[e]	A, B, C	E, F	A, B, C, D	A, B, C, D	B, C	A, B, C, D	A, B, C, D	A, B, C, D, E	A, B, C, D, E

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a. Existing facilities. All others are new or modified facilities.
b. Mechanical dilution is by either chop and dilute or press and dilute.
c. F Canyon is not required for Glass Material Oxidation and Dissolution System (GMODS) and Plasma Arc Treatment processes.
d. F Canyon is required to dissolve and dilute SNF before vitrification.
e. Fuel types: A - uranium and thorium metal fuels, B - Materials Test Reactor-like fuel, C - highly enriched uranium/low enriched uranium oxides and silicides requiring resizing or special packaging, D - loose uranium oxide in cans, E - higher-actinide targets, F - non-aluminum-clad fuels.

APPENDIX E

ASSUMED SPENT NUCLEAR FUEL MANAGEMENT ACTIVITY DURATIONS FOR ENVIRONMENTAL IMPACT ANALYSIS

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APPENDIX E. ASSUMED SPENT NUCLEAR FUEL MANAGEMENT ACTIVITY DURATIONS FOR ENVIRONMENTAL IMPACT ANALYSIS

This appendix presents the assumed durations for each spent nuclear fuel management activity necessary to implement the alternatives described in Chapter 2. DOE used these assumed durations to analyze the environmental impacts of these activities, as described in Chapter 4. These durations are used in calculating the total impacts for the following technical disciplines: worker and public health, waste generation, and utilities and energy consumption. To estimate total impacts, DOE identified the activities (phases) necessary to implement each technology, the amount of time required for each phase of the technology, and the annual impacts estimated to occur from each phase. DOE summed the annual impacts over the entire duration of each phase of a particular technology to determine the impacts of each phase, then summed the impacts of all the phases needed to implement that technology.

In estimating these durations, DOE assumed that implementation of the spent nuclear fuel management activity began in 1998 and that the final phase would end in 2035 (for a 38-year period of analysis). The years in which each technology is likely to be available are listed in Chapter 2. DOE assumed that wet storage would continue through the date that the technology was available. The projected environmental impacts of the treatment options would (on an annual basis) be greater than continued storage; assuming that wet storage would end when treatment became available is conservative. For Conventional Processing, 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. These durations (McWhorter 1997) are expressed in terms of "dissolver-years" (i.e., the time it would take to process a given fuel group using only a single canyon dissolver). However, DOE might choose to process a given fuel group using two dissolvers concurrently. In that case, the actual duration would be half that listed in McWhorter (1997), but the annual environmental impact was estimated to be twice that of a single dissolver.

For every other technology (other than Continued Wet Storage), DOE used engineering judgment to estimate the duration of the treatment phase needed to treat the entire inventory for which the technology is applicable. DOE assumed that each new treatment technology would be capable of treating the entire spent nuclear fuel inventory in 7 years based on best engineering judgements of treatment rates. Similarly, DOE assumed that the activities necessary under the Direct Disposal/Direct Co-Disposal and Repackage and Prepare to Ship technologies (characterization, fuel conditioning, cropping, etc.) also would take 7 years for the entire inventory. DOE then assumed that the fraction of the total duration attributable to each fuel type would be equal to the fraction of that fuel type's fissile mass to the total fissile mass of spent nuclear fuel in the scope of this EIS. Use of fissile mass to calculate relative treatment durations is appropriate because it approximates the total radioactivity for each fuel group. Table E-1 lists these fissile mass fractions. Tables E-2 through E-9 list the assumed durations for each phase of the eight technologies analyzed in this EIS.

After treatment, DOE assumed that the treated and packaged fuel would be put in dry storage for the duration of the 38-year period of analysis.

Table E-1. Percent of total fissile mass attributable to each fuel type[a].

Fuel group	Percent of total fissile mass (%)
A. Uranium and thorium metal fuels	1.5
B. Material test reactor-like fuels	70
C. HEU/LEU oxides and silicides requiring resizing or special packaging	19
D. Loose uranium oxide in cans	4
E. Higher actinide targets	0.6
F. Non-aluminum-clad fuels	5

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a. Source: Bickford et al. (1997).

Table E-2. Durations for Prepare for Direct Disposal/Direct Co-Disposal technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)[a]	Dry storage duration (years)
A. Uranium and thorium metal fuels	10	0.11	27.9
B. Material test reactor-like fuels	10	5.43	22.6
C. HEU/LEU oxides and silicides requiring resizing or special packaging	10	1.46	26.5
D. Loose uranium oxide in cans	NA	NA	NA
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.
a. Activities performed to prepare the fuel for direct disposal/direct co-disposal.

Table E-3. Durations for Repackage and Prepare to Ship technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)[a]	Dry storage duration (years)
A. Uranium and thorium metal fuels	NA	NA	NA
B. Material test reactor-like fuels	NA	NA	NA
C. HEU/LEU oxides and silicides requiring resizing or special packaging	NA	NA	NA
D. Loose uranium oxide in cans	NA	NA	NA
E. Higher actinide targets	10	0.04	28
F. Non-aluminum-clad fuels	10	0.35	27.65

NA = Technology is not applicable to this fuel type.
a. Activities performed to prepare the fuel for offsite shipment.

Table E-4. Durations for Melt and Dilute technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)	Dry storage duration (years)
A. Uranium and thorium metal fuels	10	0.11	27.9
B. Material test reactor-like fuels	10	5.2	22.8
C. HEU/LEU oxides and silicides requiring re-sizing or special packaging	10	1.39	26.6
D. Loose uranium oxide in cans	10	0.29	27.7
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.

Table E-5. Durations for Mechanical Dilution technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)	Dry storage duration (years)
A. Uranium and thorium metal fuels	NA	NA	NA
B. Material test reactor-like fuels	10	5.52	22.5
C. HEU/LEU oxides and silicides requiring re-sizing or special packaging	10	1.48	26.5
D. Loose uranium oxide in cans	NA	NA	NA
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.

Table E-6. Durations for Vitrification Technologies technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)	Dry storage duration (years)
A. Uranium and thorium metal fuels	10	0.11	27.9
B. Material test reactor-like fuels	10	5.2	22.8
C. HEU/LEU oxides and silicides requiring re-sizing or special packaging	10	1.39	26.6
D. Loose uranium oxide in cans	10	0.29	27.7
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.

Table E-7. Durations for Electrometallurgical Treatment technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)	Dry storage duration (years)
A. Uranium and thorium metal fuels	10	0.11	27.9
B. Material test reactor-like fuels	10	5.2	22.8
C. HEU/LEU oxides and silicides requiring re-sizing or special packaging	10	1.39	26.6
D. Loose uranium oxide in cans	10	0.29	27.7
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.

Table E-8. Durations for Conventional Processing technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)[a,b]	Dry storage duration (years)[c]
A. Uranium and thorium metal fuels	9	0.2	1
B. Material test reactor-like fuels	9	14.9	1
C. HEU/LEU oxides and silicides requiring re-sizing or special packaging	9	7.5	1
D. Loose uranium oxide in cans	9	2.2	1
E. Higher actinide targets	NA	NA	NA
F. Non-aluminum-clad fuels	NA	NA	NA

NA = Technology is not applicable to this fuel type.

a. Durations represent active processing time and do not include downtimes normally associated with processing activities.

b. Duration assumes only a single dissolver is used. If two dissolvers were used, the duration would be decreased by one-half.

c. Indicates storage of resulting low enriched uranium awaiting sale.

Table E-9. Durations for Continued Wet Storage technology.

Fuel group	Wet storage duration (years)	Treatment duration (years)	Dry storage duration (years)
A. Uranium and thorium metal fuels	38	NA	NA
B. Material test reactor-like fuels	38	NA	NA
C. HEU/LEU oxides and silicides requiring re-sizing or special packaging	38	NA	NA
D. Loose uranium oxide in cans	38	NA	NA
E. Higher actinide targets	38	NA	NA
F. Non-aluminum-clad fuels	38	NA	NA

NA = Not applicable.

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APPENDIX F

ESTIMATED INCREMENTAL NONRADIOLOGICAL AIR CONCENTRATIONS ATTRIBUTABLE TO SPENT NUCLEAR FUEL MANAGEMENT ACTIVITIES

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Table F-1. Estimated maximum incremental concentrations of nonradiological air pollutants for noninvolved worker - Uranium and Thorium Metal Fuels (Fuel Group A).[a]

Pollutant	Averaging time	Regulatory standard[b]	Incremental concentration for technology option[c]							
			1	2	3	4	5	6	7	8
Toxic pollutants (mg/m^3)										
Nitric acid	24-hour	5	<0.01	NA	<0.01	NA	2.61	<0.01	2.61	<0.01
1,1,1-trichloroethane	24-hour	1,900	–	NA	–	NA	0.02	–	0.02	–
Benzene	24-hour	3.19	–	NA	–	NA	0.02	–	0.02	–
Ethanolamine	24-hour	6	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	435	–	NA	–	NA	<0.01	–	<0.01	–
Ethylene glycol	24-hour	None	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	0.75	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	80	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	0.2	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	1,800	<0.01	NA	<0.01	NA	0.02	<0.01	0.02	<0.01
Manganese	24-hour	5	–	NA	–	NA	<0.01	–	<0.01	–
Mercury	24-hour	0.1	–	NA	–	NA	–	–	<0.01	–
Methyl alcohol	24-hour	260	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	590	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	410	–	NA	–	NA	<0.01	–	<0.01	–
Methylene chloride	24-hour	86.7	–	NA	–	NA	0.02	–	0.02	–
Napthalene	24-hour	50	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	19	–	NA	–	NA	<0.01	–	<0.01	–
Phosphorus	24-hour	0.1	–	NA	–	NA	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	2.0	–	NA	–	NA	<0.01	–	<0.01	–
Toluene	24-hour	754	<0.01	NA	<0.01	NA	0.02	<0.01	0.02	<0.01
Trichloroethene	24-hour	537	–	NA	–	NA	<0.01	–	<0.01	–
Vinyl acetate	24-hour	None	–	NA	–	NA	<0.01	–	<0.01	–
Xylene	24-hour	435	<0.01	NA	<0.01	NA	0.03	<0.01	0.03	<0.01
Criteria pollutants (μg/m^3)										
Nitrogen oxides	Annual	NA	0.02	NA	0.02	NA	36.34	0.02	36.34	–
Total suspended particulates (total dust)	24-hour	15	<0.01	NA	<0.01	NA	0.33	<0.01	0.33	–
Particulate matter (respirable fraction)	Annual	5	0.03	NA	0.03	NA	0.02	0.03	0.02	–
	24-hour	NA	0.33	NA	0.33	NA	0.20	0.33	0.20	–
Carbon monoxide	8-hour	55	0.08	NA	0.12	NA	1.57	0.12	1.57	<0.01
	1-hour	NA	0.26	NA	0.39	NA	4.89	0.39	4.89	<0.01
Sulfur dioxide	Annual	NA	<0.01	NA	<0.01	NA	0.02	<0.01	0.02	–
	8-hour	13	<0.01	NA	<0.01	NA	0.28	<0.01	0.28	–
	3-hour	NA	<0.01	NA	<0.01	NA	0.68	<0.01	0.68	–
Gaseous fluorides	1-month	None	–	NA	–	NA	0.10	–	0.10	–
	1-week	NA	–	NA	–	NA	0.17	–	0.17	–
	24-hour	NA	–	NA	–	NA	0.52	–	0.52	–
	12-hour	NA	–	NA	–	NA	0.76	–	0.76	–
Ozone (as VOC)	1-hour	0.2	NC	NA	NC	NA	NC	NC	NC	–

NA = Technology is not applicable to this fuel type.
– = No air emission associated with this option.
NC = Not Calculated.
VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. NIOSH (1991) and OSHA TWAs.

c. Technology options: 1 = Prepare for Direct Disposal/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; and 8 = Continued Wet Storage.

Table F-2. Estimated maximum incremental concentrations of nonradiological air pollutants for noninvolved worker - Materials Test Reactor-Like Fuels (Fuel Group B).[a]

	Averaging	Regulatory	Incremental concentration for technology option[c]							
Pollutant	time	standard[b]	1	2	3	4	5	6	7	8
Toxic pollutants (mg/m^3)										
Nitric acid	24-hour	5	<0.01	NA	<0.01	<0.01	3.91	<0.01	3.91	<0.01
1,1,1-trichloroethane	24-hour	1,900	–	NA	–	–	0.02	–	0.02	–
Benzene	24-hour	3.19	–	NA	–	–	0.03	–	0.03	–
Ethanolamine	24-hour	6	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	435	–	NA	–	–	<0.01	–	<0.01	–
Ethylene glycol	24-hour	None	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	0.75	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	80	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	0.2	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	1,800	<0.01	NA	<0.01	<0.01	0.03	<0.01	0.03	<0.01
Manganese	24-hour	5	–	NA	–	–	<0.01	–	<0.01	–
Mercury	24-hour	0.1	–	NA	–	–	–	–	<0.01	–
Methyl alcohol	24-hour	260	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	590	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	410	–	NA	–	–	<0.01	–	<0.01	–
Methylene chloride	24-hour	86.7	–	NA	–	–	0.03	–	0.03	–
Napthalene	24-hour	50	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	19	–	NA	–	–	<0.01	–	<0.01	–
Phosphorus	24-hour	0.1	–	NA	–	–	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	2.0	–	NA	–	–	<0.01	–	<0.01	–
Toluene	24-hour	754	<0.01	NA	<0.01	<0.01	0.03	<0.01	0.03	<0.01
Trichloroethene	24-hour	537	–	NA	–	–	<0.01	–	<0.01	–
Vinyl acetate	24-hour	None	–	NA	–	–	<0.01	–	<0.01	–
Xylene	24-hour	435	<0.01	NA	<0.01	<0.01	0.05	<0.01	0.05	<0.01
Criteria pollutants (µg/m^3)										
Nitrogen oxides	Annual	NA	0.02	NA	0.04	0.02	54.51	0.04	54.51	–
Total suspended particulates (total dust)	24-hour	15	<0.01	NA	<0.01	<0.01	0.49	<0.01	0.49	–
Particulate matter (respirable fraction)	Annual	5	0.04	NA	0.04	0.04	0.03	0.04	0.03	–
	24-hour	NA	0.49	NA	0.49	0.49	0.30	0.49	0.30	–
Carbon monoxide	8-hour	55	0.12	NA	0.19	0.12	2.35	0.19	2.35	<0.01
	1-hour	NA	0.39	NA	0.58	0.39	7.34	0.58	7.34	<0.01
Sulfur dioxide	Annual	NA	<0.01	NA	<0.01	<0.01	0.04	<0.01	0.04	–
	8-hour	13	<0.01	NA	<0.01	<0.01	0.42	<0.01	0.42	–
	3-hour	NA	<0.01	NA	<0.01	<0.01	1.02	<0.01	1.02	–
Gaseous fluorides	1-month	None	–	NA	–	–	0.14	–	0.14	–
	1-week	NA	–	NA	–	–	0.26	–	0.26	–
	24-hour	NA	–	NA	–	–	0.78	–	0.78	–
	12-hour	NA	–	NA	–	–	1.14	–	1.14	–
Ozone (as VOC)	1-hour	0.2	NC	NA	NC	NC	NC	NC	NC	–

NA = Technology is not applicable to this fuel type.
– = No air emission associated with this option.
NC = Not Calculated.
VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. NIOSH (1991) and OSHA TWAs.

c. Technology options: 1 = Prepare for Direct Disposal/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; and 8 = Continued Wet Storage.

Table F-3. Estimated maximum incremental concentrations of nonradiological air pollutants for noninvolved worker - HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (Fuel Group C).[a]

	Averaging	Regulatory	Incremental concentration for technology option[c]							
Pollutant	time	standard[b]	1	2	3	4	5	6	7	8
Toxic pollutants (mg/m^3)										
Nitric acid	24-hour	5	<0.01	NA	<0.01	<0.01	1.30	<0.01	1.30	<0.01
1,1,1-trichloroethane	24-hour	1,900	–	NA	–	–	<0.01	–	<0.01	–
Benzene	24-hour	3.19	–	NA	–	–	<0.01	–	<0.01	–
Ethanolamine	24-hour	6	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	435	–	NA	–	–	<0.01	–	<0.01	–
Ethylene glycol	24-hour	None	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	0.75	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	80	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	0.2	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	1,800	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	5	–	NA	–	–	<0.01	–	<0.01	–
Mercury	24-hour	0.1	–	NA	–	–	–	–	<0.01	–
Methyl alcohol	24-hour	260	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	590	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	410	–	NA	–	–	<0.01	–	<0.01	–
Methylene chloride	24-hour	86.7	–	NA	–	–	<0.01	–	<0.01	–
Napthalene	24-hour	50	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	19	–	NA	–	–	<0.01	–	<0.01	–
Phosphorus	24-hour	0.1	–	NA	–	–	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	2.0	–	NA	–	–	<0.01	–	<0.01	–
Toluene	24-hour	754	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	537	-	NA	-	-	<0.01	–	<0.01	–
Vinyl acetate	24-hour	None	-	NA	-	-	<0.01	–	<0.01	–
Xylene	24-hour	435	<0.01	NA	<0.01	<0.01	0.02	<0.01	0.02	<0.01
Criteria pollutants (µg/m^3)										
Nitrogen oxides	Annual	NA	<0.01	NA	0.01	<0.01	18.17	0.01	18.17	–
Total suspended particulates (total dust)	24-hour	15	<0.01	NA	<0.01	<0.01	0.16	<0.01	0.16	–
Particulate matter (respirable fraction)	Annual	5	0.01	NA	0.01	0.01	<0.01	0.01	<0.01	–
	24-hour	NA	0.16	NA	0.16	0.16	0.10	0.16	0.10	–
Carbon monoxide	8-hour	55	0.04	NA	0.06	0.04	0.78	0.06	0.78	<0.01
	1-hour	NA	0.13	NA	0.19	0.13	2.45	0.19	2.45	<0.01
Sulfur dioxide	Annual	NA	<0.01	NA	<0.01	<0.01	0.01	<0.01	0.01	–
	8-hour	13	<0.01	NA	<0.01	<0.01	0.14	<0.01	0.14	–
	3-hour	NA	<0.01	NA	<0.01	<0.01	0.34	<0.01	0.34	–
Gaseous fluorides	1-month	None	–	NA	–	–	0.05	–	0.05	–
	1-week	NA	–	NA	–	–	0.09	–	0.09	–
	24-hour	NA	–	NA	–	–	0.26	–	0.26	–
	12-hour	NA	–	NA	–	–	0.38	–	0.38	–
Ozone (as VOC)	1-hour	0.2	NC	NA	NC	NC	NC	NC	NC	–

NA = Technology is not applicable to this fuel type.
– = No air emission associated with this option.
HEU = highly enriched uranium.
LEU = low enriched uranium.
NC = Not Calculated.
VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. NIOSH (1991) and OSHA TWAs.

c. Technology options: 1 = Prepare for Direct Disposal/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; and 8 = Continued Wet Storage.

Table F-4. Estimated maximum incremental concentrations of nonradiological air pollutants for noninvolved worker - Loose Uranium Oxide in Cans (Fuel Group D).[a]

Pollutant	Averaging time	Regulatory standard[b]	Incremental concentration for technology option[c]							
			1	2	3	4	5	6	7	8
Toxic pollutants (mg/m^3)										
Nitric acid	24-hour	5	NA	NA	<0.01	NA	0.13	<0.01	0.13	<0.01
1,1,1-trichloroethane	24-hour	1,900	NA	NA	–	NA	<0.01	–	<0.01	–
Benzene	24-hour	3.19	NA	NA	–	NA	<0.01	–	<0.01	–
Ethanolamine	24-hour	6	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	435	NA	NA	–	NA	<0.01	–	<0.01	–
Ethylene glycol	24-hour	None	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	0.75	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	80	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	0.2	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	1,800	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	5	NA	NA	–	NA	<0.01	–	<0.01	–
Mercury	24-hour	0.1	NA	NA	–	NA	–	–	<0.01	–
Methyl alcohol	24-hour	260	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	590	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	410	NA	NA	–	NA	<0.01	–	<0.01	–
Methylene chloride	24-hour	86.7	NA	NA	–	NA	<0.01	–	<0.01	–
Napthalene	24-hour	50	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	19	NA	NA	–	NA	<0.01	–	<0.01	–
Phosphorus	24-hour	0.1	NA	NA	–	NA	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	2.0	NA	NA	–	NA	<0.01	–	<0.01	–
Toluene	24-hour	754	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	537	NA	NA	–	NA	<0.01	–	<0.01	–
Vinyl acetate	24-hour	None	NA	NA	–	NA	<0.01	–	<0.01	–
Xylene	24-hour	435	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Criteria pollutants (μg/m^3)										
Nitrogen oxides	Annual	NA	NA	NA	<0.01	NA	1.82	<0.01	1.82	–
Total suspended particulates (total dust)	24-hour	15	NA	NA	<0.01	NA	0.02	<0.01	0.02	–
Particulate matter (respirable fraction)	Annual	5	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	–
	24-hour	NA	NA	NA	0.02	NA	0.01	0.02	0.01	–
Carbon monoxide	8-hour	55	NA	NA	<0.01	NA	0.08	<0.01	0.08	<0.01
	1-hour	NA	NA	NA	0.02	NA	0.24	0.02	0.24	<0.01
Sulfur dioxide	Annual	NA	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	–
	8-hour	13	NA	NA	<0.01	NA	0.01	<0.01	0.01	–
	3-hour	NA	NA	NA	<0.01	NA	0.03	<0.01	0.03	–
Gaseous fluorides	1-month	None	NA	NA	–	NA	<0.01	–	<0.01	–
	1-week	NA	NA	NA	–	NA	<0.01	–	<0.01	–
	24-hour	NA	NA	NA	–	NA	0.03	–	0.03	–
	12-hour	NA	NA	NA	–	NA	0.04	–	0.04	–
Ozone (as VOC)	1-hour	0.2	NA	NA	NC	NA	NC	NC	NC	–

NA = Technology is not applicable to this fuel type.
– = No air emission associated with this option.
NC = Not Calculated.
VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. NIOSH (1991) and OSHA TWAs.

c. Technology options: 1 = Prepare for Direct Disposal/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; and 8 = Continued Wet Storage.

Table F-5. Estimated maximum incremental concentrations of nonradiological air pollutants for noninvolved worker - Higher Actinide Targets (Fuel Group E).[a]

	Averaging	Regulatory	Incremental concentration for technology option[c]							
Pollutant	time	standard[b]	1	2	3	4	5	6	7	8
Toxic pollutants (mg/m^3)										
Nitric acid	24-hour	5	NA	–	NA	NA	NA	NA	NA	<0.01
1,1,1-Trichloroethane	24-hour	1,900	NA	–	NA	NA	NA	NA	NA	–
Benzene	24-hour	3.19	NA	–	NA	NA	NA	NA	NA	–
Ethanolamine	24-hour	6	NA	–	NA	NA	NA	NA	NA	<0.01
Ethyl benzene	24-hour	435	NA	–	NA	NA	NA	NA	NA	–
Ethylene glycol	24-hour	None	NA	–	NA	NA	NA	NA	NA	<0.01
Formaldehyde	24-hour	0.75	NA	–	NA	NA	NA	NA	NA	<0.01
Glycol ethers	24-hour	80	NA	–	NA	NA	NA	NA	NA	<0.01
Hexachloronaphthalene	24-hour	0.2	NA	–	NA	NA	NA	NA	NA	<0.01
Hexane	24-hour	1,800	NA	–	NA	NA	NA	NA	NA	<0.01
Manganese	24-hour	5	NA	–	NA	NA	NA	NA	NA	–
Mercury	24-hour	0.1	NA	–	NA	NA	NA	NA	NA	–
Methyl alcohol	24-hour	260	NA	–	NA	NA	NA	NA	NA	<0.01
Methyl ethyl ketone	24-hour	590	NA	–	NA	NA	NA	NA	NA	<0.01
Methyl isobutyl ketone	24-hour	410	NA	–	NA	NA	NA	NA	NA	–
Methylene chloride	24-hour	86.7	NA	–	NA	NA	NA	NA	NA	–
Napthalene	24-hour	50	NA	–	NA	NA	NA	NA	NA	<0.01
Phenol	24-hour	19	NA	–	NA	NA	NA	NA	NA	–
Phosphorus	24-hour	0.1	NA	–	NA	NA	NA	NA	NA	–
Sodium hydroxide	24-hour	2.0	NA	–	NA	NA	NA	NA	NA	–
Toluene	24-hour	754	NA	–	NA	NA	NA	NA	NA	<0.01
Trichloroethene	24-hour	537	NA	–	NA	NA	NA	NA	NA	–
Vinyl acetate	24-hour	None	NA	–	NA	NA	NA	NA	NA	–
Xylene	24-hour	435	NA	–	NA	NA	NA	NA	NA	<0.01
Criteria pollutants (μg/m^3)										
Nitrogen oxides	Annual	NA	NA	–	NA	NA	NA	NA	NA	–
Total suspended particulates (total dust)	24-hour	15	NA	–	NA	NA	NA	NA	NA	–
Particulate matter (respirable fraction)	Annual	5	NA	–	NA	NA	NA	NA	NA	–
	24-hour	NA	NA	–	NA	NA	NA	NA	NA	–
Carbon monoxide	8-hour	55	NA	–	NA	NA	NA	NA	NA	<0.01
	1-hour	NA	NA	–	NA	NA	NA	NA	NA	<0.01
Sulfur dioxide	Annual	NA	NA	–	NA	NA	NA	NA	NA	–
	8-hour	13	NA	–	NA	NA	NA	NA	NA	–
	3-hour	NA	NA	–	NA	NA	NA	NA	NA	–
Gaseous fluorides	1-month	None	NA	–	NA	NA	NA	NA	NA	–
	1-week	NA	NA	–	NA	NA	NA	NA	NA	–
	24-hour	NA	NA	–	NA	NA	NA	NA	NA	–
	12-hour	NA	NA	–	NA	NA	NA	NA	NA	–
Ozone (as VOC)	1-hour	0.2	NA	–	NA	NA	NA	NA	NA	–

NA = Technology is not applicable to this fuel type.
– = No air emission associated with this option.
NC = Not Calculated.
VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. NIOSH (1991) and OSHA TWAs.

c. Technology options: 1 = Prepare for Direct Disposal/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; and 8 = Continued Wet Storage.

Table F-6. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for Uranium and Thorium Metal Fuels (Fuel Group A).[a]

	Averaging	Regulatory	Incremental concentration for technology option[b]							
Pollutant	time	standard	1	2	3	4	5	6	7	8
Toxic pollutants (mg/m^3)										
Nitric acid	24-hour	125	–	NA	–	NA	0.10	–	0.10	–
1,1,1-trichloroethane	24-hour	9,550	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Benzene	24-hour	150	–	NA	–	NA	<0.01	–	<0.01	–
Ethanolamine	24-hour	200	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	–	NA	–	NA	<0.01	–	<0.01	–
Ethylene glycol	24-hour	650	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	15	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	1	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	900	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	25	–	NA	–	NA	<0.01	–	<0.01	–
Mercury	24-hour	0.25	–	NA	–	NA	–	–	<0.01	–
Methyl alcohol	24-hour	1,310	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	–	NA	–	NA	<0.01	–	<0.01	–
Methylene chloride	24-hour	8,750	–	NA	–	NA	<0.01	–	<0.01	–
Napthalene	24-hour	1,250	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	–	NA	–	NA	<0.01	–	<0.01	–
Phosphorus	24-hour	0.5	–	NA	–	NA	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	50	–	NA	–	NA	<0.01	–	<0.01	–
Toluene	24-hour	2,000	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	6,750	–	NA	–	NA	<0.01	–	<0.01	–
Vinyl acetate	24-hour	176	–	NA	–	NA	<0.01	–	<0.01	–
Xylene	24-hour	4,350	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Criteria pollutants (μg/m^3)										
Nitrogen oxides	Annual	100	<0.01	NA	<0.01	NA	1.10	<0.01	1.10	<0.01
Total suspended particulates (total dust)	Annual	75	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Particulate matter (respirable fraction)	Annual	50	–	NA	–	NA	<0.01	–	<0.01	–
	24-hour	150	–	NA	–	NA	0.04	–	0.04	–
Carbon monoxide	8-hour	10,000	0.02	NA	0.03	NA	0.43	0.03	0.43	<0.01
	1-hour	40,000	0.12	NA	0.18	NA	3.20	0.18	3.20	<0.01
Sulfur dioxide	Annual	80	<0.01	NA	<0.01	NA	<0.01	<0.01	<0.01	–
	8-hour	365	<0.01	NA	0.01	NA	0.04	0.01	0.04	–
	3-hour	1,300	–	NA	–	NA	0.32	–	0.32	–
Gaseous fluorides	1-month	0.8	–	NA	–	NA	<0.01	–	<0.01	–
	1-week	1.6	–	NA	–	NA	0.01	–	0.01	–
	24-hour	2.9	–	NA	–	NA	0.02	–	0.02	–
	12-hour	3.7	–	NA	–	NA	0.04	–	0.04	–
Ozone (as VOC)	1-hour	235	0.05	NA	0.07	NA	0.26	0.08	0.26	–

NA = Technology is not applicable to this fuel type.
- = No air emission associated with this option.
+ = No state standard.
VOC = volatile organic compound.

[a] Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. Technology options: 1 = Prepare for Direct Disposal/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; and 8 = Continued Wet Storage.

Table F-7. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for Materials Test Reactor-Like Fuels (Fuel Group B).[a]

Pollutant	Averaging time	Regulatory standard	Incremental concentration for technology option[b]							
			1	2	3	4	5	6	7	8
Toxic pollutants (mg/m^3)										
Nitric acid	24-hour	125	–	NA	–	–	0.15	–	0.15	–
1,1,1-trichloroethane	24-hour	9,550	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Benzene	24-hour	150	–	NA	–	–	<0.01	–	<0.01	–
Ethanolamine	24-hour	200	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	–	NA	–	–	<0.01	–	<0.01	–
Ethylene glycol	24-hour	650	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	15	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	1	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	900	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	25	–	NA	–	–	<0.01	–	<0.01	–
Mercury	24-hour	0.25	–	NA	–	–	–	–	<0.01	–
Methyl alcohol	24-hour	1310	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	–	NA	–	–	<0.01	–	<0.01	–
Methylene chloride	24-hour	8,750	–	NA	–	–	<0.01	–	<0.01	–
Napthalene	24-hour	1,250	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	–	NA	–	–	<0.01	–	<0.01	–
Phosphorus	24-hour	0.5	–	NA	–	–	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	50	–	NA	–	–	<0.01	–	<0.01	–
Toluene	24-hour	2,000	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	6,750	–	NA	–	–	<0.01	–	<0.01	–
Vinyl acetate	24-hour	176	–	NA	–	–	<0.01	–	<0.01	–
Xylene	24-hour	4,350	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Criteria pollutants (µg/m^3)										
Nitrogen oxide	Annual	100	<0.01	NA	<0.01	<0.01	1.65	<0.01	1.65	<0.01
Total suspended particulates (total dust)	Annual	75	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Particulate matter (respirable fraction)	Annual	50	–	NA	–	–	<0.01	–	<0.01	–
	24-hour	150	–	NA	–	–	0.06	–	0.06	–
Carbon monoxide	8-hour	10,000	0.03	NA	0.05	0.03	0.65	0.05	0.65	<0.01
	1-hour	40,000	0.18	NA	0.27	0.18	4.80	0.27	4.80	<0.01
Sulfur dioxide	Annual	80	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	–
	8-hour	365	0.01	NA	0.02	0.01	0.06	0.02	0.06	–
	3-hour	1,300	–	NA	–	–	0.48	–	0.48	–
Gaseous fluorides	1-month	0.8	–	NA	–	–	<0.01	–	<0.01	–
	1-week	1.6	–	NA	–	–	0.02	–	0.02	–
	24-hour	2.9	–	NA	–	–	0.03	–	0.03	–
	12-hour	3.7	–	NA	–	–	0.06	–	0.06	–
Ozone (as VOC)	1-hour	235	0.08	NA	0.11	0.08	0.39	0.11	0.39	–

NA = Technology is not applicable to this fuel type.
– = No air emission associated with this option.
+ = No state standard.
VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. Technology options: 1 = Prepare for Direct Disposal/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; and 8 = Continued Wet Storage.

Table F-8. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging (Fuel Group C).[a]

Pollutant	Averaging time	Regulatory standard	Incremental concentration for technology option[b]							
			1	2	3	4	5	6	7	8
Toxic pollutants (mg/m^3)										
Nitric acid	24-hour	125	–	NA	–	–	0.05	–	0.05	–
1,1,1-trichloroethane	24-hour	9,550	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Benzene	24-hour	150	–	NA	–	–	<0.01	–	<0.01	–
Ethanolamine	24-hour	200	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	–	NA	–	–	<0.01	–	<0.01	–
Ethylene glycol	24-hour	650	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	15	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	1	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	900	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	25	–	NA	–	–	<0.01	–	<0.01	–
Mercury	24-hour	0.25	–	NA	–	–	–	–	<0.01	–
Methyl alcohol	24-hour	1310	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	–	NA	–	–	<0.01	–	<0.01	–
Methylene chloride	24-hour	8,750	–	NA	–	–	<0.01	–	<0.01	–
Napthalene	24-hour	1,250	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	–	NA	–	–	<0.01	–	<0.01	–
Phosphorus	24-hour	0.5	–	NA	–	–	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	20	–	NA	–	–	<0.01	–	<0.01	–
Toluene	24-hour	2,000	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	6,750	–	NA	–	–	<0.01	–	<0.01	–
Vinyl acetate	24-hour	176	–	NA	–	–	<0.01	–	<0.01	–
Xylene	24-hour	4,350	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Criteria pollutants ($\mu g/m^3$)										
Nitrogen oxide	Annual	100	<0.01	NA	<0.01	<0.01	0.55	<0.01	0.55	<0.01
Total suspended particulates (total dust)	Annual	75	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
Particulate matter (respirable fraction)	Annual	50	–	NA	–	–	<0.01	–	<0.01	–
	24-hour	150	–	NA	–	–	0.02	–	0.02	–
Carbon monoxide	8-hour	10,000	0.01	NA	0.02	0.01	0.22	0.02	0.22	<0.01
	1-hour	40,000	0.06	NA	0.09	0.06	1.60	0.09	1.60	<0.01
Sulfur dioxide	Annual	80	<0.01	NA	<0.01	<0.01	<0.01	<0.01	<0.01	–
	8-hour	365	<0.01	NA	<0.01	<0.01	0.02	<0.01	0.02	–
	3-hour	1,300	–	NA	–	–	0.16	–	0.16	–
Gaseous fluorides	1-month	0.8	–	NA	–	–	<0.01	–	<0.01	–
	1-week	1.6	–	NA	–	–	<0.01	–	<0.01	–
	24-hour	2.9	–	NA	–	–	0.01	–	0.01	–
	12-hour	3.7	–	NA	–	–	0.02	–	0.02	–
Ozone (as VOC)	1-hour	235	0.03	NA	0.04	0.03	0.13	0.04	0.13	–

NA = Technology is not applicable to this fuel type.
– = No air emission associated with this option.
+ = No state standard.
HEU = highly enriched uranium.
LEU = low enriched uranium.
VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. Technology options: 1 = Prepare for Direct Disposal/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; and 8 = Continued Wet Storage.

Table F-9. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for Loose Uranium Oxide in Cans (Fuel Group D).[a]

	Averaging	Regulatory	Incremental concentration for technology option[b]							
Pollutant	time	standard	1	2	3	4	5	6	7	8
Toxic pollutants (mg/m^3)										
Nitric acid	24-hour	125	NA	NA	–	NA	<0.01	–	<0.01	–
1,1,1-trichloroethane	24-hour	9,550	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Benzene	24-hour	150	NA	NA	–	NA	<0.01	-	<0.01	-
Ethanolamine	24-hour	200	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Ethyl benzene	24-hour	4,350	NA	NA	–	NA	<0.01	-	<0.01	-
Ethylene glycol	24-hour	650	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Formaldehyde	24-hour	7.5	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Glycol ethers	24-hour	+	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexachloronaphthalene	24-hour	1	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Hexane	24-hour	200	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Manganese	24-hour	25	NA	NA	–	NA	<0.01	–	<0.01	-
Mercury	24-hour	0.25	NA	NA	–	NA	–	–	<0.01	–
Methyl alcohol	24-hour	1,310	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl ethyl ketone	24-hour	14,750	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Methyl isobutyl ketone	24-hour	2,050	NA	NA	–	NA	<0.01	–	<0.01	–
Methylene chloride	24-hour	8,750	NA	NA	–	NA	<0.01	–	<0.01	–
Napthalene	24-hour	1,250	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Phenol	24-hour	190	NA	NA	–	NA	<0.01	–	<0.01	–
Phosphorus	24-hour	0.5	NA	NA	–	NA	<0.01	–	<0.01	–
Sodium hydroxide	24-hour	20	NA	NA	–	NA	<0.01	–	<0.01	–
Toluene	24-hour	2,000	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Trichloroethene	24-hour	6,750	NA	NA	–	NA	<0.01	–	<0.01	–
Vinyl acetate	24-hour	176	NA	NA	–	NA	<0.01	–	<0.01	–
Xylene	24-hour	4,350	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Criteria pollutants (μg/m^3)										
Nitrogen oxide	Annual	100	NA	NA	<0.01	NA	0.06	<0.01	0.06	<0.01
Total suspended particulates (total dust)	Annual	75	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	<0.01
Particulate matter (respirable fraction)	Annual	50	NA	NA	–	NA	<0.01	–	<0.01	–
	24-hour	150	NA	NA	–	NA	<0.01	–	<0.01	–
Carbon monoxide	8-hour	10,000	NA	NA	<0.01	NA	0.02	<0.01	0.02	<0.01
	1-hour	40,000	NA	NA	<0.01	NA	0.16	<0.01	0.16	<0.01
Sulfur dioxide	Annual	80	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	–
	8-hour	365	NA	NA	<0.01	NA	<0.01	<0.01	<0.01	–
	3-hour	1,300	NA	NA	–	NA	0.02	–	0.02	–
Gaseous fluorides	1-month	0.8	NA	NA	–	NA	<0.01	–	<0.01	–
	1-week	1.6	NA	NA	–	NA	<0.01	–	<0.01	–
	24-hour	2.9	NA	NA	–	NA	<0.01	–	<0.01	–
	12-hour	3.7	NA	NA	–	NA	<0.01	–	<0.01	–
Ozone (as VOC)	1-hour	235	NA	NA	<0.01	NA	0.01	<0.01	0.01	–

NA = Technology is not applicable to this fuel type.
– = No air emission associated with this option.
+ = No state standard.
VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. Technology options: 1 = Prepare for Direct Disposal/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; and 8 = Continued Wet Storage.

Table F-10. Estimated maximum incremental concentrations of nonradiological air pollutants at SRS boundary for Higher Actinide Targets (Fuel Group E).[a]

	Averaging	Regulatory	Incremental concentration for technology option[b]							
Pollutant	time	standard	1	2	3	4	5	6	7	8
Toxic pollutants (mg/m^3)										
Nitric acid	24-hour	125	NA	–	NA	NA	NA	NA	NA	–
1,1,1-trichloroethane	24-hour	9,550	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Benzene	24-hour	150	NA	-	NA	NA	NA	NA	NA	–
Ethanolamine	24-hour	200	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Ethyl benzene	24-hour	4,350	NA	-	NA	NA	NA	NA	NA	-
Ethylene glycol	24-hour	650	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Formaldehyde	24-hour	15	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Glycol ethers	24-hour	+	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Hexachloronaphthalene	24-hour	1	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Hexane	24-hour	900	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Manganese	24-hour	25	NA	–	NA	NA	NA	NA	NA	–
Mercury	24-hour	0.25	NA	–	NA	NA	NA	NA	NA	–
Methyl alcohol	24-hour	1,310	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Methyl ethyl ketone	24-hour	14,750	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Methyl isobutyl ketone	24-hour	2,050	NA	–	NA	NA	NA	NA	NA	–
Methylene chloride	24-hour	515	NA	–	NA	NA	NA	NA	NA	–
Napthalene	24-hour	1,250	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Phenol	24-hour	190	NA	–	NA	NA	NA	NA	NA	–
Phosphorus	24-hour	0.5	NA	–	NA	NA	NA	NA	NA	–
Sodium hydroxide	24-hour	50	NA	–	NA	NA	NA	NA	NA	–
Toluene	24-hour	2,000	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Trichloroethene	24-hour	6,750	NA	–	NA	NA	NA	NA	NA	–
Vinyl acetate	24-hour	176	NA	–	NA	NA	NA	NA	NA	–
Xylene	24-hour	4,350	NA	<0.01	NA	NA	NA	NA	NA	<0.01
Criteria pollutants (μg/m^3)										
Nitrogen oxide	Annual	100	NA	–	NA	NA	NA	NA	NA	<0.01
Total suspended particulates (total dust)	Annual	75	NA	–	NA	NA	NA	NA	NA	<0.01
Particulate matter (respirable fraction)	Annual	50	NA	–	NA	NA	NA	NA	NA	–
	24-hour	150	NA	–	NA	NA	NA	NA	NA	–
Carbon monoxide	8-hour	10,000	NA	–	NA	NA	NA	NA	NA	<0.01
	1-hour	40,000	NA	–	NA	NA	NA	NA	NA	<0.01
Sulfur dioxide	Annual	80	NA	–	NA	NA	NA	NA	NA	–
	8-hour	365	NA	–	NA	NA	NA	NA	NA	–
	3-hour	1,300	NA	–	NA	NA	NA	NA	NA	–
Gaseous fluorides	1-month	0.8	NA	–	NA	NA	NA	NA	NA	–
	1-week	1.6	NA	–	NA	NA	NA	NA	NA	–
	24-hour	2.9	NA	–	NA	NA	NA	NA	NA	–
	12-hour	3.7	NA	–	NA	NA	NA	NA	NA	–
Ozone (as VOC)	1-hour	245	NA	<0.01	NA	NA	NA	NA	NA	

NA = Technology is not applicable to this fuel type.
– = No air emission associated with this option.
+ = No state standard.
VOC = volatile organic compound.

a. Not all constituents listed in this table appear in Tables 3.3-3 or 3.3-4 because many constituents are not expected to impact SRS ambient air concentrations.

b. Technology options: 1 = Prepare for Direct Disposal/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; and 8 = Continued Wet Storage.

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United States Senate

The Honorable Ernest F. Hollings
United States Senate

The Honorable Strom Thurmond
United States Senate

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The Honorable Robert C. Byrd
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Chairman
Subcommittee on Energy and Water Development
Committee on Appropriations

The Honorable Carl Levin
Ranking Minority Member
Committee on Armed Services

The Honorable Harry Reid
Ranking Minority Member
Subcommittee on Energy and Water Development
Committee on Appropriations

The Honorable Robert Smith
Chairman
Subcommittee on Strategic Forces
Committee on Armed Services

The Honorable Ted Stevens
Chairman
Committee on Appropriations

The Honorable John Warner
Chairman
Committee on Armed Services

A.3 UNITED STATES HOUSE OF REPRESENTATIVES FROM AFFECTED AND ADJOINING STATES

The Honorable James E. Clyburn
U.S. House of Representatives

The Honorable Nathan Deal
U.S. House of Representatives

The Honorable Lindsey Graham
U.S. House of Representatives

The Honorable Jack Kingston
U.S. House of Representatives

The Honorable Cynthia McKinney
U.S. House of Representatives

The Honorable Charlie Norwood
U.S. House of Representatives

The Honorable Mark Sanford
U.S. House of Representatives

The Honorable Floyd Spence
U.S. House of Representatives

The Honorable John M. Spratt, Jr.
U.S. House of Representatives

A.4 UNITED STATES HOUSE OF REPRESENTATIVES COMMITTEES

The Honorable Peter Visclosky
Ranking Minority Member
Subcommittee on Energy and Water
Development
Committee on Appropriations

The Honorable Duncan L. Hunter
Chairman
Subcommittee on Military Procurement
Committee on National Security

The Honorable C. W. Bill Young
Chairman
Committee on Appropriations

The Honorable Ron Packard
Chairman
Subcommittee on Energy and Water
Development
Committee on Appropriations

The Honorable David Obey
Ranking Minority Member
Committee on Appropriations

The Honorable Norman Sisisky
Ranking Minority Member
Subcommittee on Military Procurement
Committee on National Security

The Honorable Ike Skelton
Ranking Minority Member
Committee on Armed Services

The Honorable Floyd Spence
Chairman
Committee on National Security

B. FEDERAL AGENCIES

Mr. John Bellinger
NEPA Coordinator
Office of Environmental Policy
U.S. Army Corps of Engineers

Major General R. M. Bunker
Division Engineer, South Atlantic Division
U.S. Army Corps of Engineers

Mr. Bill Champion
West Valley Nuclear Services
U.S. Department of Energy

Mr. Douglas H. Chapin
Richland Operations Office
U.S. Department of Energy

Mr. Ken Clark
Region II Public Affairs Officer
U.S. Nuclear Regulatory Commission
Commander, Savannah District
Attn: Planning Division
U.S. Army Corps of Engineers

Ms. Marjorie S. Davenport
District Chief
Water Resources Division
U.S. Geological Survey
U.S. Department of Interior

Mr. Paul F. X. Dunigan, Jr.
Richland Operations Office
U.S. Department of Energy

Mr. Leif Erickson
Director
Richland Operations Office
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Mr. Robert Fairweather
Chief, Environmental Branch
Office of Management and Budget

Dr. Haus Mark
Assistant Secretary for Nuclear, Chemical and
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Mr. Joseph R. Franzmathes
Assistant Regional Administrator, Office of Policy and Management
U.S. Environmental Protection Agency, Region IV

Mr. Mark W. Frei
Office of Project Completion
Deputy Assistant Secretary
U.S. Department of Energy

Mr. Kenneth W. Holt
Centers for Disease Control and Prevention
National Center for Environmental Health
U.S. Department of Health and Human Services

Mr. Dave Huizenga
Office of Integration and Disposition
Deputy Assistant Secretary
U.S. Department of Energy

Mr. Michael Jansky
WM Hanford
U.S. Department of Energy

Mr. Don L. Klima
Director, Office of Planning & Review
Advisory Council on Historic Prevention

Mr. Andreas Mager, Jr.
Habitat Conservation Division
National Marine Fisheries Service
National Oceanic and Atmospheric Administration
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Mr. Heinz Mueller
Office of Environmental Policy & Compliance
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Mr. Charles Oravetz
Chief
Protected Species Management Branch
Southeast Regional Office
National Marine Fisheries Service
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Mr. Carl J. Paperiello
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U.S. Department of Energy Laboratory

Mr. Jon Richards
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Mr. Willie R. Taylor
Director
Office of Environmental Policy & Compliance
U.S. Department of Interior

Mr. Greer C. Tidwell
Administrator, US EPA-IV

Ms. Kathy Whitaker
Idaho Operations Office
U.S. Department of Energy

C. STATE OF SOUTH CAROLINA

C.1 STATEWIDE OFFICES AND LEGISLATURE

The Honorable Jim Hodges
Governor of South Carolina

The Honorable Bob Peeler
Lieutenant Governor of South Carolina

The Honorable Charles Condon
Attorney General

The Honorable James E. Smith, Jr.
South Carolina House of Representatives

Ms. Omeaga Burgess
Office of the State Budget

C.2 STATE AND LOCAL AGENCIES AND OFFICIALS

The Honorable Jackie Holman
Mayor of Blackville

Coordinator
Aiken County Civil Defense
Aiken County Emergency Services
Attn: Freddie M. Bell

Mr. Russell Berry
South Carolina Department of Health and Environmental Control

Mr. Frank Brafman
Hilton Head Town Council

Mr. Donnie Cason
South Carolina Department of Highways and Public Transportation

Ms. Ann Clark
Federal Facility Liaison
Environmental Quality Control
South Carolina Department of Health and Environmental Control

Mr. Phil England
Director
Aiken County Planning & Development Department

Mr. G. Kendall Taylor
Division of Hydrogeology
Bureau of Land and Hazardous Waste Management
South Carolina Department of Health and Environmental Control

Mr. David Wilson
Division of Hydrogeology
Bureau of Land and Hazardous Waste Management
South Carolina Department of Health and Environmental Control

D. STATE OF GEORGIA

D.1 STATEWIDE OFFICES AND LEGISLATURE

The Honorable Roy Barnes
Governor of Georgia

The Honorable Mark Taylor
Lieutenant Governor of Georgia

The Honorable Charles W. Walker
Georgia Senate

The Honorable Thurbert Baker
Attorney General

The Honorable Ben L. Harbin
Georgia House of Representatives

E. NATURAL RESOURCE TRUSTEES, SAVANNAH RIVER SITE

Mr. Douglas E. Bryant
Commissioner, SCDHEC
Natural Resource Trustee

Mr. A. B. Gould
Director
Natural Resource Trustee
DOE-SR Environmental Quality Management

Mr. Clarence Ham
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US Army Corps of Engineers
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Mr. David Holroyd
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Mr. Ronald W. Kinney
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Mr. James Setser
Chief, Program Coordinator Branch
SRS Natural Resource Trustee
Department of Natural Resources

Dr. Paul A. Sandifer
Director
SC Department of Natural Resources
SRS Natural Resource Trustee

F. NATIVE AMERICAN GROUPS

The Honorable Gilbert Blue
Chairman
Catawba Indian Nation

The Honorable Bill Fife
Principal Chief
Muscogee (Creek) Nation

G. ENVIRONMENTAL AND PUBLIC INTEREST GROUPS

Mr. Bill Cunningham
Economist
Department of Public Policy
AFL-CIO

Ms. Susan Gordon
Program Director
Alliance for Nuclear Accountability

Mr. Brad Morse
Alliance for Nuclear Accountability

Ms. Karen Patterson
SRS Citizens Advisory Board

Ms. Elizabeth Hoinkes
General Counsel
Arms Control and Disarmament Agency

Dr. Alan Kuperman
Brookings Institute

Mr. Bennett Ramberg
Committee to Bridge the Gap

Dr. Arjun Makhijani
President
Institute for Energy and Environmental Research

Mr. Brian Costner
Institute for Energy and Environmental Research

Dr. Mildred McClain
Citizens for Environmental Justice, Inc.

Mr. Paul Schwartz
National Campaigns Director
Clean Water Action

Mr. William Yeniscavich
Defense Nuclear Facilities Safety Board

Mr. Andrew Thibadeau
Information Officer
Defense Nuclear Facilities Safety Board

Mr. Jim Melillo
Executive Director
Environmental Management Advisory Board

Dr. Brent Blackwelder
President
Friends of the Earth

Mr. Tom Clements
Executive Director
Nuclear Control Institute

Mr. David Moglen
Greenpeace
Washington, D.C.

Dr. David Albright
President
Institute for Science and International Security

Dr. David Bradley
National Community Action Foundation

Mr. Robert Holden
Executive Director
National Congress of American Indians

Mr. Alex Echols
Deputy Director
National Fish and Wildlife Foundation

Ms. Tamar Osterman
Director of Government Affairs
Department of Law & Public Policy
National Trust for Historic Preservation

Mr. Thomas F. Donnelly
Executive Vice President
National Water Resources Association

Mr. Mark Van Putten
President & Chief Executive Officer
National Wildlife Foundation

Dr. Thomas V. Cochran
Director, Nuclear Programs
Natural Resources Defense Council

Dr. Paul Leventhal
President, Nuclear Control Institute

Dr. Ed Lyman
Scientific Director
Nuclear Control Institute

Mr. Bob Tiller
Director of Security Programs
Physicians for Social Responsibility

Mr. David Culp
Plutonium Challenge

Ms. Joy Oakes
Regional Staff Director
Appalachian Office
The Sierra Club

Mr. David Becker
The Sierra Club

Mr. Tom Zamora Collina
Director of Arms Control Project
Union of Concerned Scientists

Ms. Diane Jackson
Administrative Assistant
Ecology and Economics Research Department
The Wilderness Society

Ms. Rebecca Charles
Tennessee Department of Environment and Conservation

H. OTHER GROUPS AND INDIVIDUALS

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Dr. Dave Amick
SAIC

Mr. Tom Anderson
Battelle

Ms. Margaret Aoki
NINNHO IWAI American Corporation

Ms. Jila Banaee
Lockheed-Martin Idaho Technologies Company

Mr. Sy Baron
MUSC

Ms. Sonya Barnette

Mr. James R. Barrett
B&W Services, Inc.

Colonel Mary Barton

Mr. Robert C. Baumback

Ms. Lisa Baxter
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Mr. W. Alton Bell, Jr.
Plumbers & Steamfitters Local Union #150

Mr. Charles T. Black

Mr. Edward P. Blanton, Jr.

Mr. Edmund D. Boothe
Aiken Technical College

R. P. Borsody

Mr. Carlos W. Bowen

Ms. Sara Jo Braid

Mr. Dannion Brinkley

Mr. Bill Brizes

Ms. Elizabeth R. Brown
Charleston Deanery
South Carolina Council of Catholic Women

Mr. Ethan Brown

Mr. Ken Bulmahn

Ms. Donna Campbell
Foster Wheeler Environmental Corporation

Mr. Rich Campbell
Chem-Nuclear Systems

Mr. Roy Carter

Mr. George R. Caskey

Dr. Kailash Chandra

Mr. Ernie Chaput

Mr. Vladimir Y. Chechik
Shaw, Pitman, Potts, & Trowbridge

Mr. Carl E. Cliche

Ms. Marlena Conde
Edlow International Company

Mr. Steve Connor

Mr. S. W. Corbett

Mr. Stephen Crump

Dr. Tim Devol
Clemson University
Environmental Systems Engineering Department

Mr. Sal Dimaria

Mr. John Dimarzio

Mr. John F. Doherty, J.D.

Mr. Dave Ecklund

Mr. Richard Engelmann
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Mr. Ken Fitch

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Mr. Melvyn P. Galin

Mr. Ben Gannon

Mr. John Geddie

Colonel George A. Gibson

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Mr. Robert Guild

Mr. Jan Hagers

Mr. David Haines

Ms. Mary Hassell

Ms. Kathryn Hauer

Ms. Cathy Haupt

Ms. Shelley Hawkins
Jacobs Engineering Group, Inc.

Mr. Warren Hills
Laborers Local 1137
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Mr. C. C. Holcomb

Mr. Robert A. Hollingsworth

Mr. Leonard Huesties

Mr. Charles E. Irvin

Mr. Cliff Jarman

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Mr. Norman Kaish

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Mr. Larry Kripps

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Miss Cynthia E. Lake

Mr. Dewey E. Large
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Mr. Lechel

Mr. Thomas L. Lippert

Dr. William A. Lochstet
University of Pittsburgh at Johnstown Physics

Mr. Alastair J. MacDonald
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Mr. Robert Maher

Mr. Steve Maheras

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ES&H Program Control Manager
Argonne Medical Laboratory-West

Mr. Joseph A. Martillotti
Texas Department of Health
Bureau of Radiation Control

Mr. Bob Matthews

Ms. Elizabeth McBride

Ms. Trish McCracken

Dr. William R. McDonell

Mr. Michael F. McGowan
Geological Environmental Consultant

Ms. Dana McIntyre
WJBF-TV
Channel 6

Mr. Frank Metz

Ms. Louise M. Montgomery

Mr. Emment Moore
Washington State University

Mr. Kenneth J. Newcomer

Ms. Kim Newell
Public Information Director
SCDHEC

Mr. R. I. Newman

Mr. David Nichols

Ms. Suzanne Ochse

Mr. Matthew J. O’Connor

Mr. James L. Oliver

Mr. J. F. Ortaldo

Mr. Robert F. Overman

Mr. Aris Papadopoulous

Ms. Jean Pasquale

Dr. Ruth Patrick
Division of Limnology and Ecology
Academy of Natural Sciences of Philadelphia

Mr. Mark A. Petermann
Hydrogeologist
RMT, Inc.

Mr. Jeff Petraglia

Mr. George Piper

Mr. David Pittman

Mr. W. Lee Poe

Mr. Ron Pound
Environmental Projects
Newsletter

Mr. Richard H. Powell

Ms. Essie M. Richards
Carver Heights Community Org.

Ms. Dorene L. Richardson

Mr. Wayne Rickman
Sonalysts, Inc.

Mr. F. A. Riemann

Mr. Jacob Charles Robertson

Mr. Paul Robinson
Southwest Research & Information Center

Ms. Linda Rodgers

Ms. Connie Rogers

Mr. Edward Scalsky

Mr. Glenn R. Schlafer

Mr. Guy R. Selph

Dr. R. F. Shangraw, Jr.
Project Performance Corporation

Ms. Margaret Shekell
Ultra Systems Environmental

Mr. John O. Shipman

Ms. Kimberly Sizemore

Mr. Jim Skinner

Mr. Donald J. Skinner

Mr. Arthur H. Smith, Jr.

Mr. Don Solki
Carpenter's Local 283

Mr. Paul Stansbury
PNL

Mr. Jim Steinke
Newport News Shipbuilding Co.

Mr. Bill Stokes
Advanced Nuclear and Medical Systems

The Reverend Thomas A. Summers

Mr. Arthur Sutherland
Rogers & Associates Engineering Corporation

Mr. Edward S. Syrjala

Dr. D. William Tedder
Associate Professor
School of Chemical Engineering
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Mr. James W. Terry
Oak Ridge National Laboratory
Lockheed Martin Energy Research

Mr. Vincent Van Brunt
University of South Carolina
Chemical Engineering Department

Ms. Linda Van Sickle
Exploration Resources

Mr. Alan Vaughan
Nuclear Fuel Services

Mr. Martin Vorum
Commodore Advanced Sciences, Inc.

Ms. Melissa Vrana
Project Performance Corporation

Mr. Jim Wanzeck

Mr. Payton H. Ward, Jr.
Ironworkers Local Union #709

Mr. Edgar West
Ironworkers Local Union #709

Mr. Frank S. Watters

Mr. Kim Welsch

Dr. F. Ward Whicker
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Ms. Reba White
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Ms. Pam Whitson
Oak Ridge Associated Universities

Mr. Patrick L. Whitworth

Mrs. Debbie Wilcox

Mr. Don J. Wilkes
Jacob Engineering Group

Mr. Jermetia L. Williams

Mr. Michael Witunski

Mr. Mel Woods

Dr. Abe Zeitoun
SAIC
Mr. Francis P. Zera
The Georgia Guardian

I. READING ROOMS AND LIBRARIES

Freedom of Information Public Document Room
University of South Carolina at Aiken SC
Gregg-Graniteville Library

Freedom of Information Reading Room
U.S. Department of Energy

Battelle-Pacific Northwest Laboratories
Attn: Technical Library

Librarian
Chatham-Effingham-Liberty Regional Library

Librarian
Los Alamos Technical Association

Head, Document Department
The Libraries
Colorado State University

Librarian
Erskine College
McCain Library

Parsons Brinckerhoff Library

Public Reading Room
Chicago Operations Office

Mr. Michael Simpson
Library of Congress

Ms. Judy Smith
Monographs Acquisition Services
Colorado State Universities Libraries

Felicia Yeh
Technical Services Librarian
South Carolina State Library

Librarian
Savannah River Site Library
Savannah River Technology Center

Librarian
Westinghouse Savannah River Company Library
776-H

Librarian
Orangeburg County Free Library

Librarian
South Carolina State Library
Columbia, SC

Librarian
Technical Library
Argonne National Laboratory

Librarian
Georgia Institute of Technology Library

Librarian
Pullen Public Library
Atlanta, GA

Librarian
Freedom of Information Act (FOIA) Reading Room
DOE-Albuquerque Operations Office

Public Reading Room
DOE Oak Ridge Operations Office

Librarian
Reese Library
Augusta College

Librarian
County Library
Charleston, SC

Librarian
Government Information Department
Zimmerman Library
University of New Mexico

Public Reading Room
Richland, WA

Public Reading Room
Idaho National Engineering and Environmental
Laboratory

GLOSSARY

°C

Degree Celsius. A temperature scale commonly used in scientific work based on the freezing point of water at 0°C and the boiling point at 100°C under normal atmospheric pressure. $°C = \frac{5}{9} \times (°F - 32)$.

°F

Degree Fahrenheit. A temperature scale based on the freezing point of water at 32°F and the boiling point at 212°F under normal atmospheric pressure. $°F = (°C \times \frac{9}{5}) + 32$.

absorbed dose

The energy deposited per unit mass by ionizing radiation. The unit of absorbed dose is the rad.

accident

One or more unplanned events involving materials that have the potential to 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 adverse health effects.

actinide

Any of a series of chemically similar, mostly synthetic, radioactive elements with atomic numbers ranging from actinium at 89 through lawrencium at 103.

air quality standards

The prescribed level of constituents in the outside air (ambient air) that legally should not be exceeded during a specified unit of time in a specified area.

alpha (α) particle

A positively charged particle consisting of two protons and two neutrons that is emitted from the nucleus of certain nuclides during radioactive decay. It is the least penetrating of the three common types of radiation (alpha, beta, and gamma).

aqueous

Relating to or made with water.

aquifer

A geologic formation that contains enough saturated porous material to permit groundwater to move through it and to yield worthwhile quantities of groundwater to wells and springs.

As Low As Reasonably Achievable (ALARA)

An approach to radiation protection that controls or manages exposures (both individual and collective to workers and general public) as low as social, technical, economic, practical and public policy considerations permit. ALARA is not a dose limit, but a process which has the objective of dose levels as far below applicable limits of 10 CFR 835 as is reasonably achievable. Particular attention is to be paid to this definition in design of facilities.

atomic weight

The relative weight of an atom of a chemical element based on the weight of the most abundant isotope of carbon, which is taken to be 12.

AXAIRQ

A computer model that analyzes doses from airborne radionuclide releases.

background exposure

See exposure to radiation.

background radiation

Normal radiation present in the lower atmosphere from cosmic rays and earth sources. Background radiation varies with location, depending on altitude and natural radioactivity present in the surrounding geology.

Best Management Practices (BMP)

A practice or combination of practices that is determined by a state (or other planning agency) to be the most effective, practicable means of preventing pollution generated by nonpoint sources or of reducing it to a level compatible with air or water quality goals.

beta (β) particle

An elementary particle emitted from a nucleus during radioactive decay. It is negatively charged, identical to an electron, and easily stopped by a thin sheet of metal.

biota

Living organisms.

blackwater

Water in Coastal Plain creeks, swamps, and rivers that has a dark or black coloration due to dissolution of naturally-occurring organic matter from soils and decaying vegetation.

bounding accident

An accident whose calculated consequences encompass all other possible accident consequences for that facility. For example, a bounding accident for the release of hazardous material from a storage tank would postulate the release of the entire tank contents. The consequences from this accident would be greater than the consequences of all other tank release accidents.

brownfield

An area that has been previously disturbed by industrial activities.

burn

Irradiation of fuel in a nuclear reactor with the resultant production of energy, neutrons, and fission products.

burnup

The total energy released through fission by a given amount of nuclear fuel; generally measured in megawatt-days.

cancer

A malignant tumor of potentially unlimited growth, capable of invading surrounding tissue or spreading to other parts of the body by metastasis.

canister

A stainless-steel container in which nuclear material is sealed.

canyon

A heavily shielded building where radioactive materials are chemically processed to recover special isotopes for national defense or other programmatic purposes. In the canyon, operation and maintenance are remotely-controlled.

capable (geology)

Describes a geological fault that has moved at or near the ground surface within the past 35,000 years.

carcinogen

An agent capable of inducing cancer.

carcinogenic

Capable of inducing cancer.

case

The application of a given technology to a single fuel group.

cask

A massive, heavily-shielded container for holding nuclear materials during shipment.

cesium

Naturally occurring element with 55 protons in its nucleus. Some manmade isotopes of cesium are radioactive (e.g., cesium-134, cesium-137).

cladding

The outer jacket of fuel elements and targets, usually made of aluminum, stainless steel, or zirconium-aluminum alloy; used to prevent fuel corrosion and retain fission products during reactor operation, or to prevent radioactive releases into the environment during storage.

co-disposal

A disposal approach for spent nuclear fuel in a geologic repository. Five canisters of high-level waste would fit in a repository waste package, with room for one 17-inch (43-centimeter) diameter canister of spent nuclear fuel.

collective dose

The sum of the individual doses to all members of a specific population.

committed dose equivalent

The calculated dose equivalent received by a tissue or an organ during the 50-year period after a radionuclide is introduced into the body.

committed effective dose equivalent

The sum of the committed dose equivalents to various tissues/organs in the body multiplied by their appropriate tissue weighting factor. Equivalent in effect to a uniform external dose of the same value.

community (environmental justice)

A group of people or a specific location exposed to risks that potentially threaten health, ecology, or land values, or exposed to industry that stimulates unwanted noise, smell, industrial traffic, particulate matter, or other nonaesthetic impacts.

concentration

The amount of a substance contained in a unit quantity of material.

confining unit

A body of impermeable or distinctly less-permeable material stratigraphically adjacent to one or more aquifers.

consequence

The result or effect (especially projected exposure to radiological or chemical hazards) of a release of hazardous materials to the environment.

constituents

Parts or components of a whole.

critical

Describing the condition when fissile materials exposed to neutron bombardment produce enough neutrons to support a chain reaction.

criticality

A state in which a self-sustaining nuclear chain reaction is achieved.

crop

A process that cuts off or otherwise removes the hardware on the fuel assemblies, leaving primarily the active fuel for subsequent processes.

cumulative impacts

Additive impacts on the environment including ecological, human health, or socioeconomic effects that result from the addition of the impact of the proposed action to impacts from other past, present, and reasonably foreseeable future actions regardless of what agency (Federal or non-Federal) or person undertakes the other actions (40 CFR 1508.7).

curie (Ci)

A unit of radioactivity equal to 37,000,000,000 disintegrations per second (or becquerels).

daughter

A nuclide formed by the radioactive decay of another nuclide, which is the “parent.”

decay heat

The radioactive decay of fission products can produce very high temperatures (decay heat), which is why fuel recently removed from a reactor is placed in underwater storage for cooling. Without active cooling, the fuel could overheat and melt or damage the cladding. After a sufficient cooling time that depends on the burnup of the fuel and its composition, fuel assemblies can be stored dry. Dry fuel storage technologies must consider decay heat.

decay, radioactive

The spontaneous transformation of one nuclide into a different nuclide or into a different energy state of the same nuclide. The process results in the emission of nuclear radiation (usually alpha, beta, or gamma radiation).

decibel

A unit for measuring the relative loudness of sounds. In general, a sound doubles in loudness for every increase of 10 decibels.

decision maker

Group or individual responsible for making a decision on a particular proposed action. Decision makers include DOE officials as specified in DOE Order 451.1A; elected officials; Federal, state, and local agency representatives; and the public.

decommissioning

The removal from service of facilities such as processing plants, waste tanks, and burial grounds, and the reduction or stabilization of radioactive contamination. Decommissioning includes decontamination, dismantling, and return of the area to original condition without restrictions or partial decontamination, isolation of remaining residues, and continuation of surveillance and restrictions.

Defense Waste Processing Facility

Savannah River Site facility that processes high-level radioactive waste into a glass form for transport to a permanent disposal site.

deflagration

Rapid burning with great heat and intense light.

demographic

Related to the statistical study of human populations, including size, density, distribution, and vital statistics such as age, gender, and ethnicity.

depleted uranium

A mixture of uranium isotopes where uranium-235 represents less than 0.7 percent of the uranium by mass.

derived concentration guide (DCG)

The concentration of a radionuclide in air or water that, under conditions of continuous exposure for one year by one exposure mode (i.e., ingestion of water, submersion in air, or inhalation), would result in an effective dose equivalent of 100 mrem (0.1 rem = 1 mSv [milliSievert]).

disassociate

Separate chemicals into their elemental or ionic state.

dose

The energy imparted to matter by ionizing radiation. The unit of absorbed dose is the rad, equal to 0.01 joule per kilogram of irradiated material in any medium.

dose conversion factor

Factor used to calculate the dose received from exposure to radiation.

dose equivalent

A term used to express the amount of effective radiation when modifying factors have been considered. It is the product of absorbed dose (rads) multiplied by a quality factor and other modifying factors. It is measured in rem (*R*oentgen *e*quivalent *m*an).

dose rate

The radiation dose delivered per unit time (e.g., rem per year).

ecology

The science that deals with the relationship of living things with each other and with their environment.

ecosystem

A complex of the communities of living things and their environment which forms a functioning whole in nature.

effective dose equivalent

A quantity used to estimate the biological effect of ionizing radiation. It is the sum over all body tissues of the product of absorbed dose, the quality factor (to account for the different penetrating abilities of the various radiations), and the tissue weighting factor (to account for the different radiosensitivity of the various tissues of the body).

effective porosity

A property of earth containing interconnecting interstices, expressed as a percent of bulk volume occupied by the interstices.

effluent

Liquid or airborne material released to the environment. In general usage, however, effluent implies liquid releases.

electron

An elementary particle with a mass of 9.107×10^{-28} gram (or 1/1837 of a proton) and a negative charge. Electrons surround the positively charged nucleus and determine the chemical properties of the atom.

element

One of the 109 known chemical substances that cannot be divided into simpler substances by chemical means. All isotopes of an element have the same atomic number (number of protons) but have different numbers of neutrons.

Emergency Response Planning Guideline (ERPG) values

These values, which are specific for each chemical, are established for three general severity levels: exposure to concentrations greater than ERPG-1 values for a period of time greater than 1 hour results in an unacceptable likelihood that a person would experience mild transient adverse health effects, or perception of a clearly defined objectionable odor; exposure to concentrations greater than ERPG-2 values for a period of time greater than 1 hour results in an unacceptable likelihood that a person would experience or develop irreversible or other serious health effects, or symptoms that could impair one's ability to take protective action; exposure to concentrations greater than ERPG-3 values for a period of time greater than 1 hour results in an unacceptable likelihood that a person would experience or develop life-threatening health effects.

emission standards

Legally enforceable limits on the quantities and kinds of air contaminants that may be emitted to the atmosphere.

endangered species

Plants or animals that are in danger of extinction through all or a significant portion of their ranges and that have been listed as endangered by the U.S. Fish and Wildlife Service or the National Marine Fisheries Service.

energy

The capacity to produce heat or do work.

enrichment

A process in which the fraction of the uranium-235 isotope has been artificially increased above the natural abundance level of 0.72 percent.

environment

The sum of all external conditions and influences affecting the life, development, and ultimately the survival of an organism.

environmental impact statement (EIS)

A detailed written statement as required by Section 102(2)(c) of the National Environmental Policy Act (NEPA) of 1969, as amended, to assess the environmental impacts of major Federal actions.

environmental justice

The fair treatment of people of all races, cultures, incomes, and educational levels with respect to the development, implementation, and enforcement of environmental laws, regulations, and policies. Fair treatment implies that no population of people should be forced to shoulder a disproportionate share of the negative environmental impacts of pollution or environmental hazards due to a lack of political or economic strength.

exposure to radiation

The incidence of radiation on living or inanimate material by accident or intent. Background exposure is the exposure to natural background ionizing radiation. Occupational exposure is the exposure to ionizing radiation that occurs at a person's workplace. Population exposure is the exposure to a number of persons who inhabit an area.

external initiators

Accidental occurrences that 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.

fault

A fracture or a zone of fractures within a rock formation along which vertical, horizontal, or transverse slippage of the earth's crust has occurred in the past.

fertile

Describing radionuclides that can be converted into fissile material (e.g., thorium-232 and uranium-238 can be converted through neutron capture to uranium-233 and plutonium-239, respectively).

fissile

Capable of being split or divided (fissioned) by the absorption of thermal neutrons. The most common fissile materials are uranium-233, uranium-235, and plutonium-239.

fission

The splitting of a heavy nucleus into two approximately equal parts, which are nuclei of lighter elements, accompanied by the release of energy and generally two or more neutrons. Fission can occur spontaneously or can be induced by nuclear bombardment.

fission chain reaction

Nuclear reaction in which atomic nuclei in reactor fuel respond to collisions with neutrons by splitting into two or three major fragments and additional neutrons accompanied by the emission of gamma radiation.

fission fragments

The parts into which atomic nuclei in reactor fuel split during a fission chain reaction.

fission products

Nuclei from the fission of heavy elements (primary fission products); also, the nuclei formed by the decay of the primary fission products, many of which are radioactive.

gamma (γ) rays

High-energy, short-wavelength electromagnetic radiation accompanying fission, radioactive decay, or nuclear reactions. Gamma rays are very penetrating and require relatively thick shields to absorb the rays effectively.

geology

The science that deals with the earth: the materials, processes, environments, and history of the planet, especially the lithosphere, including the rocks, their formation and structure.

groundwater

The supply of fresh water below the earth's surface in an aquifer.

habitat

The place or type of site where a plant or animal normally grows or lives.

half-life (radiological)

The time in which half the atoms of a radioactive substance disintegrate to another nuclear form. Half-lives vary from millionths of a second to billions of years.

hazardous material

A substance or a material including a hazardous substance that has been determined by the U.S. Secretary of Transportation to be capable of posing an unreasonable risk to health, safety, and property when transported in commerce.

hazardous substance

Any substance that when released to the environment in an uncontrolled fashion could be harmful to the biota or human health and when released in an unpermitted fashion becomes subject to the reporting and possible response provisions of the Clean Water Act and the Comprehensive Environmental Response, Compensation, and Liability Act.

hazardous waste

Waste that is regulated under the Resource Conservation and Recovery Act and corresponding state regulations. Waste is hazardous if the EPA lists it 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, chromates, lead, paint solvents, and various laboratory chemicals.

heavy metal

In this document, heavy metal refers to materials of high atomic number that were placed in nuclear reactors. This includes thorium, uranium, and plutonium.

high efficiency particulate air (HEPA) filter

A type of filter designed to remove 99.95 percent of the particles down to 0.3 micrometer in diameter from a flowing air stream.

high(ly) enriched uranium

Uranium that is equal to or greater than 20 percent uranium-235 by weight. Many of the fuels discussed in this EIS are based primarily on highly enriched uranium.

high-level radioactive waste

Highly radioactive material from the processing of spent nuclear fuel 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.

hydraulic conductivity

The rate of water flow in gallons pre day through a cross-section of 1 square foot under a unit hydraulic gradient, also known as permeability coefficient.

hydraulic gradient

With regard to an aquifer, the rate of change of pressure head per unit distance of flow at a given point and in a given direction.

infrastructure

The system of public works of a county, state, or region; also, the resources (buildings or equipment) required for an activity.

interim storage

Safe and secure storage for spent nuclear fuel and radioactive wastes until the materials are dispositioned (treatment and/or disposal).

internal initiators

Events that normally originate in and around the facility but are always a result of facility operations (equipment or structural failures, human errors, internal flooding). In accident scenarios, initiators start the events that culminate in a release of hazardous or radioactive materials.

involved worker

An individual located in the facility under discussion.

ion

An atom or molecule that has gained or lost one or more electrons to become electrically charged.

ionizing radiation

Radiation capable of ejecting electrons from atoms or molecules to produce ions.

irradiation

Exposure to radiation.

isotope

An atom of a chemical element with a specific atomic number and atomic weight. Isotopes of the same element have the same number of protons but different numbers of neutrons (i.e., the same atomic number, but different mass numbers). Isotopes are identified by the name of the element and the total number of protons and neutrons in the nucleus. For example, plutonium-239 is a plutonium atom with 239 protons and neutrons.

isotopic dilution

Mixing a less-enriched radioisotope with a highly enriched radioisotope to yield an isotope with lower nuclear enrichment.

latent cancer fatalities

Deaths resulting from cancers that became active sometime after the exposure presumed to have induced them.

long-lived radionuclides

Radioactive isotopes with half-lives greater than about 30 years.

low-enriched uranium (LEU)

Uranium with uranium-235 enriched above the natural concentration (0.72 percent) but below 20 percent; highly enriched uranium (HEU) is enriched 20 percent or higher.

low-income community

A community where 25 percent or more of the population is identified as living in poverty.

low-level mixed waste

Radioactive waste that contains material listed as hazardous under the Resource Conservation and Recovery Act or that exhibits one or more of the following hazardous waste characteristics: ignitability, corrosivity, reactivity, or toxicity. It includes such materials as tritiated mercury, tritiated oil contaminated with mercury, other mercury-contaminated compounds, or radioactively-contaminated lead shielding.

low-level radioactive waste

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.

materials test reactor equivalent (MTRE)

A quantity of spent nuclear fuel related to its volume that provides information on the amount of storage space provided.

MAXIGASP

A computer program used to calculate doses of airborne releases of radioactivity to the maximally exposed member of the public.

maximally exposed individual

A hypothetical person located to receive the maximum possible dose by a given exposure scenario.

maximum contaminant levels (MCLs)

The maximum permissible level of a contaminant in water that is delivered to a user of a public water system.

metric tons of heavy metal (MTHM)

Quantities of unirradiated and spent nuclear fuel and targets are traditionally expressed in terms of metric tons of heavy metal (typically uranium) without the inclusion of other materials such as cladding, alloy materials, and structural materials. A metric ton is 1,000 kilograms, which is equal to about 2,200 pounds.

migration

The natural travel of a material through the air, soil, or water.

millirem

One thousandth of a rem. (See rem.)

minority community

A person classified by the U.S. Bureau of the Census as Black, Hispanic, Asian and Pacific Islander, American Indian, Eskimo, Aleut, or other nonwhite persons is considered a minority. A community with the number of minority persons equal to or greater than the minority average of a defined area or jurisdiction (usually a state) is a minority community.

moderation

Process for slowing down neutrons resulting from fission or other nuclear reactions; slow or “thermal” neutrons are necessary for sustaining a fission chain reaction in fissile materials; water and heavy water are common moderators.

monitoring

Continuing control and accountability, particularly of special nuclear materials such as plutonium-239 and highly enriched uranium, but also including oversight of hazardous or reactive compounds before they are disposed of or converted to a stable long-term storage form.

National Ambient Air Quality Standards

Air quality standards established by the Clean Air Act, as amended in 1990. The primary National Ambient Air Quality Standards are intended to provide the public with an adequate margin of safety, and the secondary National Ambient Air Quality Standards are intended to protect the public from known or anticipated adverse impacts of a pollutant.

National Pollutant Discharge Elimination System

Federal system that permits liquid effluents regulated through the Clean Water Act, as amended.

natural phenomena initiators

Natural occurrences that are independent of facility operations and events at nearby facilities or operations (earthquakes, high winds, floods, lightning, snow). Although these initiators are independent of external facilities, they can affect such facilities and compound the progression of the accident.

natural radiation or natural radioactivity

Background radiation. Radiation arising from cosmic and terrestrial naturally-occurring radionuclide sources.

National Environmental Policy Act (NEPA)

The National Environmental Policy Act of 1969 (42 USC 4321) requires the preparation of an EIS for Federal projects that could incur significant impacts to the environment.

neutron

An elementary nuclear particle capable of inducing a fission chain reaction in certain atomic nuclei, including uranium-235.

neutron poison

A substance that absorbs neutrons without causing a fission, thereby preventing nuclear criticalities.

noninvolved worker

For this EIS, an SRS worker who is not involved in a given operation or activity.

nonproliferation

The restriction of access to fissile materials in concentrations sufficient to assemble a nuclear weapon.

Nuclear Regulatory Commission

The independent Federal commission that licenses and regulates commercial nuclear facilities.

nuclear radiation

Radiation, usually alpha, beta, or gamma, that emanates from an unstable atomic nucleus.

nuclear reaction

An interaction between a photon, particle, or nucleus and a target nucleus, leading to transmutation.

nuclear reactor

A device in which a fission chain reaction is maintained, used for the irradiation of materials or the generation of electricity.

nuclide

A species of atom characterized by the number of protons, number of neutrons and by energy content in the nucleus; a radionuclide is a radioactive nuclide.

offsite population

Defined as all individuals located within an 80-kilometer (50-mile) radius of a facility with potential to emit radioactive material.

organic compounds

Chemical compounds containing hydrocarbons.

ozone

A compound of oxygen in which three oxygen atoms are chemically attached to each other.

oxides of nitrogen (NO_x)

Primarily nitrogen oxide (NO) and nitrogen dioxide (NO_2), these compounds are produced in the combustion of fossil fuels, and contribute to air pollution.

particulates

Solid particles and liquid droplets small enough to become airborne.

passivation

To reduce the reactivity of a chemically-active metal.

pellets

One configuration of the reactive material in a target rod.

permeability

A measure of a material’s ability to have liquids or gases passes through it via pores or openings.

person-rem

The radiation dose to a given population; the sum of the individual doses received by a population.

plutonium (Pu)

A transuranic, heavy (average atomic mass about 244 atomic mass units), silvery metal with 15 known isotopes that is produced by the neutron irradiation of natural uranium. Plutonium-239 is used both in nuclear weapons and commercial nuclear power applications. Plutonium-238 is used to power onboard electric generators during manned and unmanned space flights.

poison

A material that has an affinity for absorbing neutrons. Poisons are added to nuclear materials with a potential criticality concern to lessen the likelihood of an uncontrolled nuclear reaction.

pollution

The addition of an undesirable agent to an ecosystem in excess of the rate at which natural processes can degrade, assimilate, or disperse it.

POPGASP

A computer model used to calculate doses of airborne releases of radioactivity to the population within 80 kilometers (50 miles) of the SRS.

population

In this EIS, a collection of members of the public that is located outside the boundaries of the SRS. Impacts in this EIS are estimated for the population within a given area, depending on the appropriate environmental pathways. For example, the affected population for liquid releases to the Savannah River includes downstream residents.

Prevention of Significant Deterioration (PSD)

A standard that establishes the acceptable amount of deterioration in air quality. When the air quality of an area meets the standards for a specific pollutant, the area is declared to be in attainment for that pollutant. When the air quality of an area does not meet the standard for a specific pollutant, the area is said to be in nonattainment for that pollutant. PSD requirements define maximum allowable increases (increments) in ambient air pollutant concentrations (sulfur dioxide, particulate, nitrogen oxide) for construction or modification of facilities which by definition do not "significantly deteriorate" the existing baseline air quality.

processing (of spent nuclear fuel)

Applying a chemical or physical process designed to alter the characteristics of the spent fuel matrix.

production well/water

At the SRS, water treated and used as potable water.

programmatic materials

Stable nuclear materials with value for supporting national programs (e.g., plutonium-238 production for the National Aeronautics and Space Administration).

proton

A nuclear particle with a positive charge equal in magnitude to the negative charge of the electron; it is a constituent of all atomic nuclei, and the atomic number of an element indicates the number of protons in the nucleus of each atom of that element.

pyrophoric

The tendency to spontaneously ignite in air. Some uranium and thorium metal fuels may be pyrophoric.

radiation

The emitted particles and photons from the nuclei of radioactive atoms; a shortened term for ionizing radiation or nuclear radiation as distinguished from nonionizing radiation (microwaves, ultraviolet rays, etc.).

radiation shielding

Radiation-absorbing material that is interposed between a source of radiation and organisms that would be harmed by the radiation (e.g., people).

radioactive waste

Waste that has radioactive material and must be handled as such.

radioactivity

The spontaneous decay of unstable atomic nuclei, accompanied by the emission of radiation.

radioisotope

Radioactive isotopes. The isotopes of an element that are radioactive. Not all isotopes of a single element are radioactive. Some radioisotopes are naturally occurring (e.g., potassium-40) while others are produced by nuclear reactions.

radiolysis

Decomposition of a material by ionizing radiation.

radionuclide

A nuclide that exhibits radioactivity.

reactor

A device in which a chain reaction of fissionable material is initiated and controlled; a nuclear reactor.

Record of Decision (ROD)

A document that provides a concise public record of an agency decision on a proposed action described in an EIS. An ROD identifies the alternatives, the environmentally preferable alternative(s), factors the agency considered in making the decision, and whether the agency has adopted all practicable means to avoid or minimize environmental harm and if not, why not.

recycling

Return of a waste material either to the process that generated the waste or to another process to use or reuse the waste material beneficially; recovery of a useful or valuable material from waste.

rem

The unit of radiation dose for biological absorption. It is equal to the product of the absorbed dose in rads, a quality factor and a distribution factor.

remote handling cell

A room designed so that the process carried out in the room is done remotely by operators manipulating robotic equipment.

repository

A place for the disposal of immobilized high-level waste and spent nuclear fuel in isolation from the environment.

processing (of spent nuclear fuel)

Processing of reactor-irradiated nuclear material (primarily spent nuclear fuel) to recover fissile and fertile material, in order to recycle such materials primarily for defense programs or generation of electricity. Historically, processing has involved aqueous chemical separations of elements (typically uranium or plutonium) from undesired elements in the fuel.

resin

An ion-exchange medium; organic polymer used for the preferential removal of certain ions from a solution.

Richter Scale

A scale to quantify earthquake intensity.

risk

In accident analysis, the probability-weighted consequence of an accident, defined as the accident frequency per year multiplied by the dose. The term “risk” also is used commonly in other applications to describe the probability of an event occurring.

road-ready

Describing spent nuclear fuel that has been conditioned or treated and placed in a canister in a form such that it can be shipped to a repository.

runoff

The portion of rainfall, melted snow, or irrigation water that flows across ground surface and eventually returns to streams. Runoff can carry pollutants into receiving waters.

saltstone

Low-radioactivity fraction of high-level waste mixed with cement, flyash, and slag to form a concrete matrix.

sanitary waste

Solid waste that is neither hazardous, as defined by the Resource Conservation and Recovery Act, nor radioactive. It consists of salvageable material and material that is suitable for disposition in a municipal sanitary landfill. Sanitary waste streams include such items as paper, glass, discarded office material, and construction debris.

seismicity

The tendency for earthquakes to occur.

shielded transport casks

A heavily shielded container designed to hold one or more fuel elements during transport.

short-lived

A designation for radionuclides with relatively short half-lives (i.e., they decay to other atoms relatively quickly). Radionuclides with half-lives less than approximately 30 years are short-lived.

spent nuclear fuel

Fuel and targets that have been irradiated in a nuclear reactor. Spent nuclear fuel is highly radioactive.

stabilization

The action of making a nuclear material more chemically or physically stable by converting its physical or chemical form or placing it in a more stable environment.

strontium

Naturally occurring element with 38 protons in its nucleus. Some manmade isotopes of strontium are radioactive (e.g., strontium-89, strontium-90).

sulfur dioxide

A heavy, pungent, toxic gas, used as a preservative or refrigerant, that is an air pollutant.

surface water

Water on the surface (streams, ponds, etc.), as distinguished from underground water.

tank farm

An installation of interconnected underground tanks for the storage of high-level radioactive liquid wastes.

target

In this EIS, a tube of material placed in a reactor for bombardment by neutrons to produce desired radioactive byproducts.

thermal neutrons

Neutrons that have had excess energy removed by scattering collisions with other atoms and will not slow down any further. Thermal neutrons have an energy of 0.025 electron-volts and are readily absorbed by fissile atoms.

threatened species

Any species which is likely to become endangered within the foreseen future throughout all or a significant portion of its range, and that has been listed as threatened by the U.S. Fish and Wildlife Service or the National Marine Fisheries Service.

transmutation

The conversion of one element to another by means of a nuclear reaction.

transuranic waste

Waste material containing more than a specified concentration of transuranic elements (elements with higher atomic numbers than uranium) (presently, more than 10 nanocuries per gram of waste).

tritium

A radioactive isotope of hydrogen; its nucleus contains one proton and two neutrons.

uranium (U)

A heavy (average atomic mass of about 238 atomic mass units), silvery-white metal with 14 radioactive isotopes. One of the isotopes, uranium-235, is most commonly used as fuel for nuclear fission and another, uranium-238, is transformed into fissionable plutonium-239 following its capture of a neutron in a nuclear reactor.

vault

A reinforced concrete structure for storing strategic nuclear materials used in national defense or other programmatic purposes or for disposing of radioactive or hazardous waste.

vitrification

Immobilization by incorporating into glass.

vulnerability

Condition or weakness that could lead to exposure to radioactive elements by the public, unnecessary or increased exposure to workers, or release of radioactive materials to the environment.

Waste Isolation Pilot Plant (WIPP)

DOE facility located near Carlsbad, New Mexico, for the safe underground disposal of transuranic waste from numerous facilities owned by DOE.

waste minimization

Reduction of waste before treatment, storage, or disposal by source reduction or recycling activities.

waste, radioactive

See “radioactive waste.”

water quality standard

Provisions of state or Federal law that consist of a designated use or uses for the waters of the United States and water quality criteria for such waters based upon those uses. Water quality standards are used to protect the public health or welfare, enhance the quality of water, and serve the purposes of the Act.

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This list names the individuals who contributed to the technical content of this environmental impact statement (EIS). The preparation of the EIS was directed by K. E. Waltzer and M. R. Woodworth of the U.S. Department of Energy (DOE) and P. L. Young and J. L. Oliver of Tetra Tech NUS, Inc.

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Subtask 09

National Environmental Policy Act (NEPA) Documentation, Management, and Planning Support. Environmental Documentation/NEPA Requirements Support for Site Specific Spent Nuclear Fuel Management Program at SRS.

I hereby certify (or as a representative of my organization, I hereby certify) that, to the best of my knowledge and belief, no facts exist relevant to any past, present, or currently planned interest or activity (financial, contractual, personal, organizational or otherwise) which relate to the proposed work and bear on whether I have (or the organization has) a possible conflict of interest with respect to (1) being able to render impartial, technically sound, and objective assistance or advice, or (2) being given an unfair */ competitive advantage.

Signature: Janet Mandel   Date: June 6, 1996

Name: Janet Mandel   Organization: Halliburton NUS Corporation

Title: Contracting Officer

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*/ An unfair competitive advantage does not include the normal flow of benefits from the performance of this contract.

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