

**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 stor-

age 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 1,000°C, 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 646°C, with melter temperature in the range 750 to 850°C 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 <sup>a</sup>
20	400
5	1,234
2	1,796

a. For representative inventory of processed aluminum-SNF assemblies, assuming 0.276 m<sup>3</sup> 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.

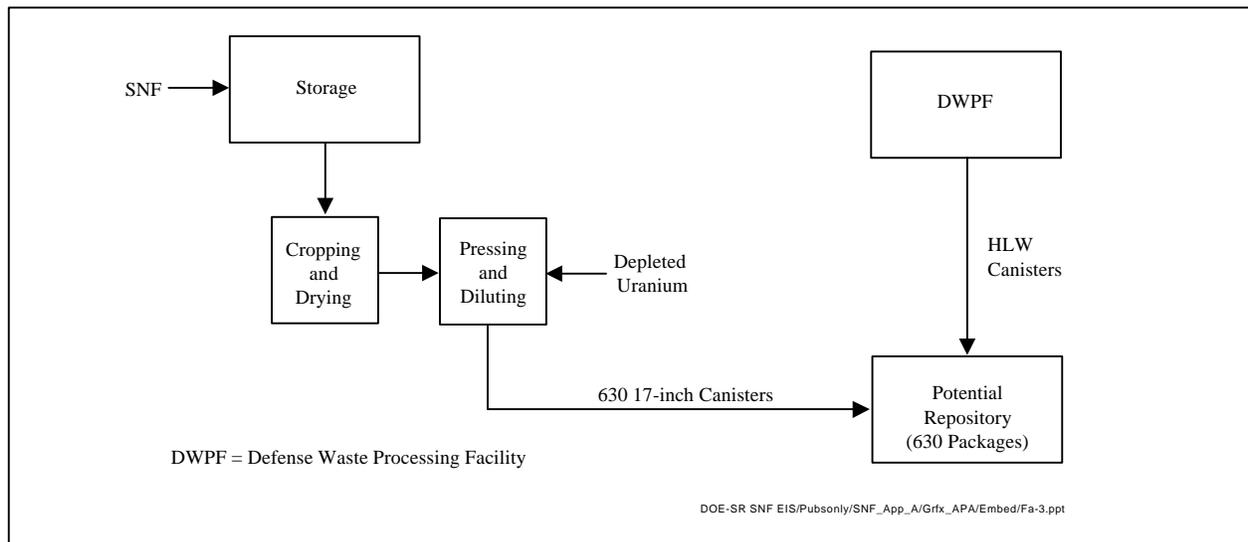
The primary advantage of Press and Dilute technology is its simplicity. However, the variable sizes of Group C SNF might make the technology unsuitable for those fuels without special disassembly before compaction. Particulate fuels (Group D) would not be amenable to pressing. Figure A-3 shows the Press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

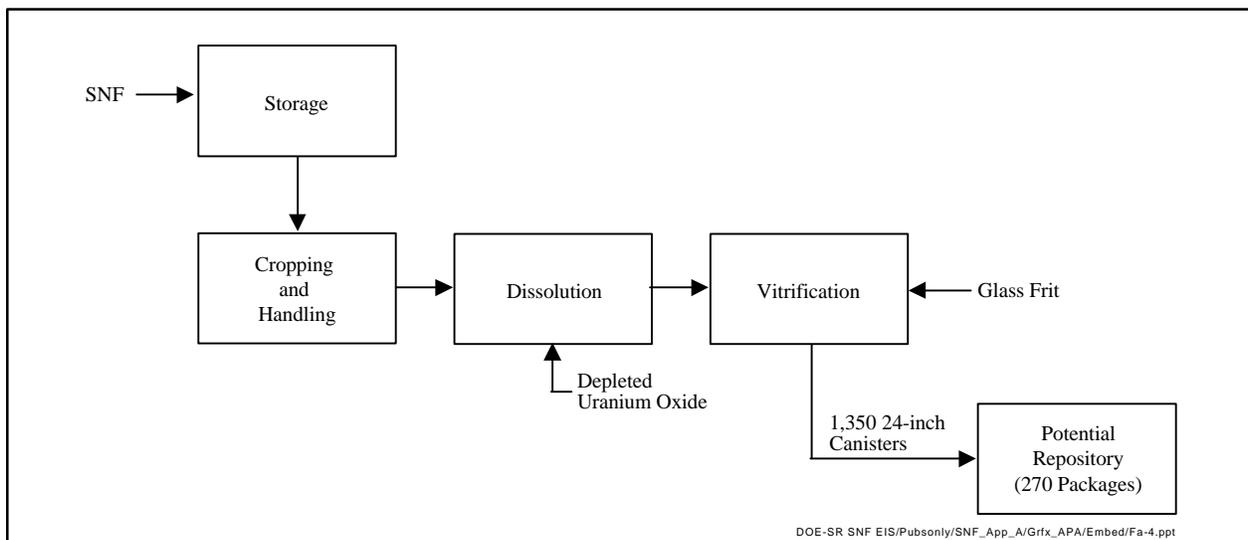


Figure A-4. Dissolve and Vitrify process flow diagram.

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

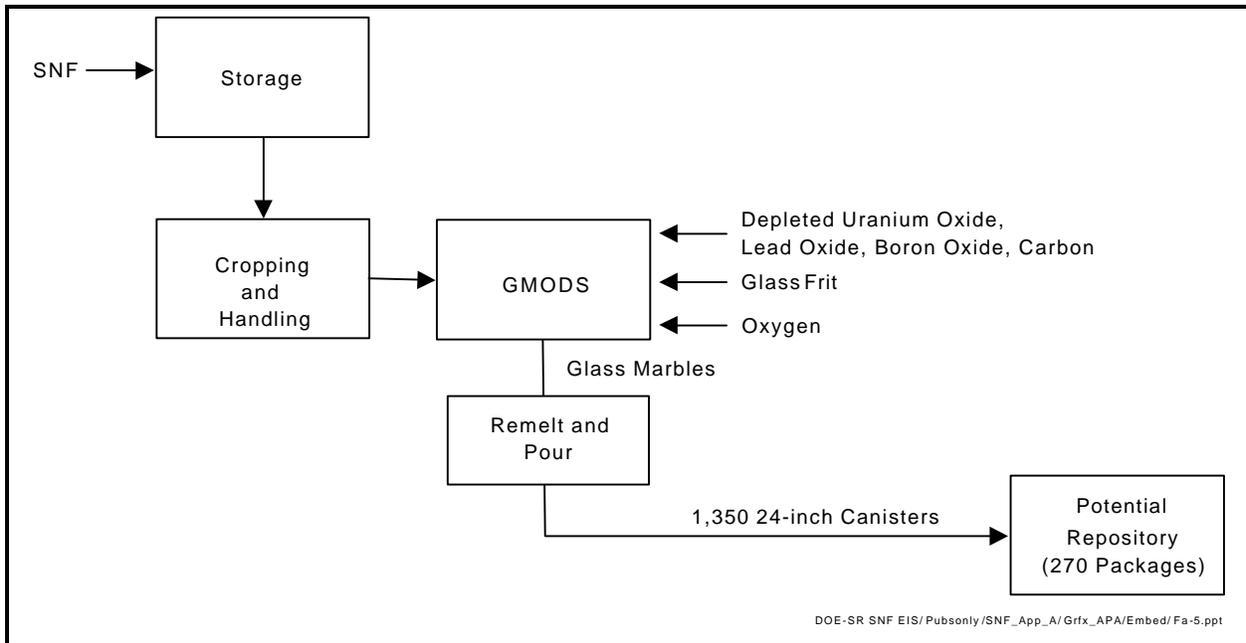
remelted, allowing a continuous pour to fill several 24-inch diameter canisters at a time. The GMODS process would produce typically about 1,350 canisters for emplacement in about 270 repository packages.

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

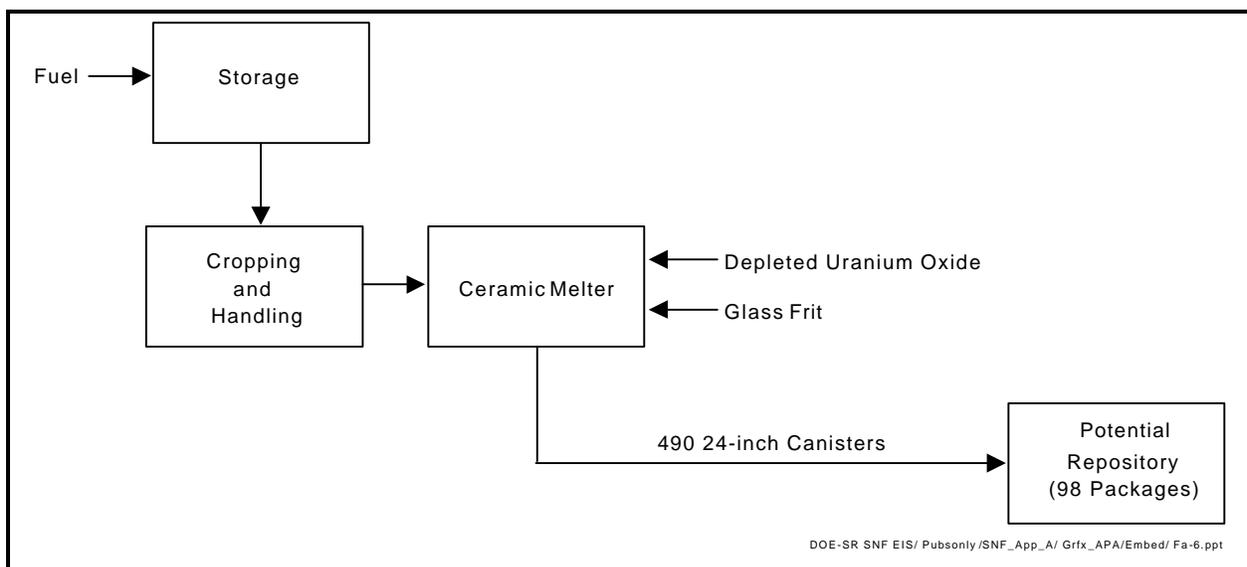
plasma arc furnace for incorporation into the ceramic product. Figure A-6 shows the Plasma Arc Treatment process flow diagram.

**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



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

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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 ura-

nium metal deposits would be collected and separated from electrolyte salts as was the aluminum.

### **Salt Scrubbing**

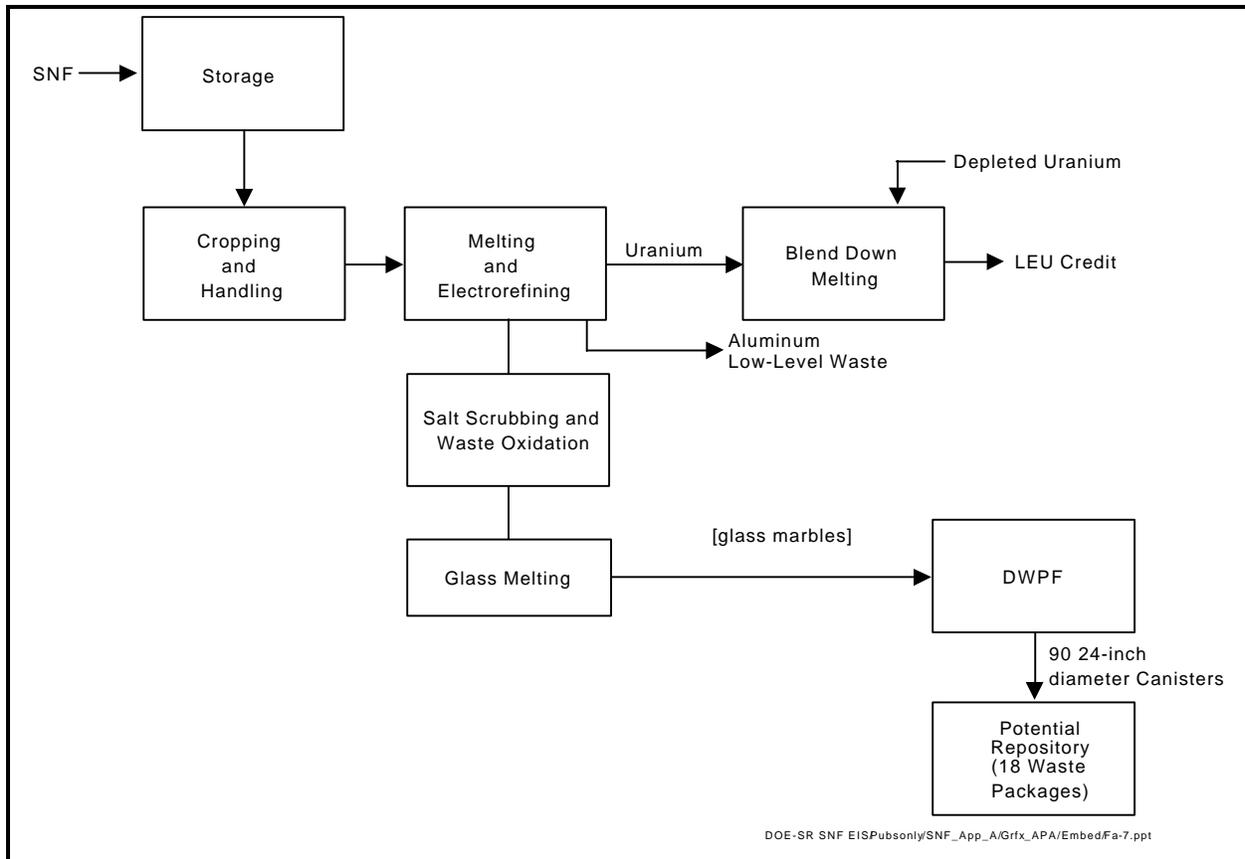
With the continued electrorefining of the SNF, alkaline earth fluorides would build up in the aluminum electrorefiner, and the rare earth and transuranic fluorides would build up in the uranium electrorefiner. These waste products could be separated from the electrolyte by ion exchange or chemical reduction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



**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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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.

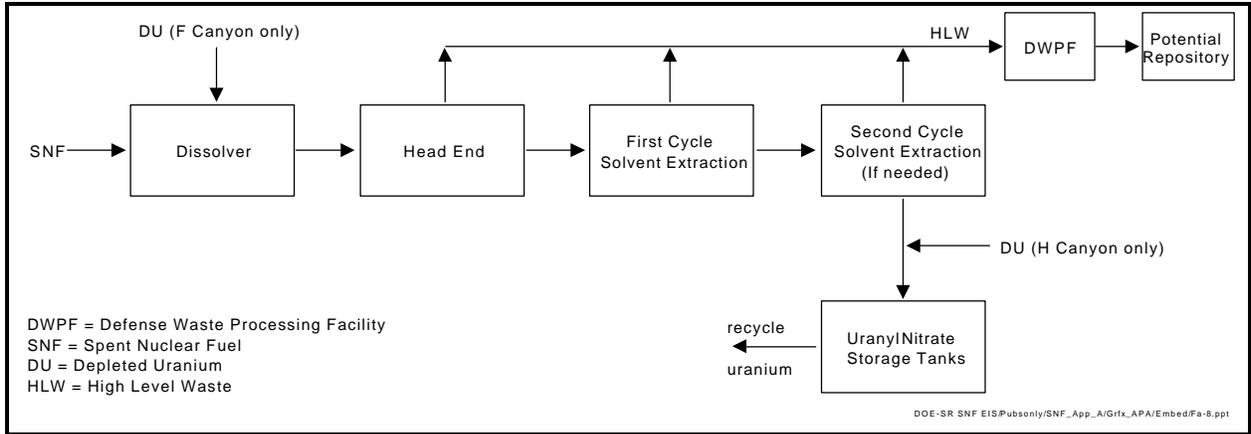
Chemical processing produces an acidic solution that is neutralized before transfer to large tanks in the F- and H-Area Tank Farms. During storage to allow short-lived radionuclides to decay, the insoluble components of the alkaline waste

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 cementitious 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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Figure A-8. Conventional Processing flow diagram.

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