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SAVANNAH RIVER SITE

PLANNED DISPOSITION OF MATERIAL IN THE SAVANNAH RIVER SITE HIGH LEVEL WASTE SYSTEM

WSRC-RP-2003-00323

March 17, 2003

Executive Summary

Since the early 1950's Savannah River Site (SRS) has received over 100 million gallons of waste into the F and H Tank Farms, commonly known as the High Level Waste System (HLW) system. This waste was neutralized, insoluble sludge settled, and supernate was evaporated to form saltcake such that as of January 2003 only 37 million gallons of waste is stored in F and H tank farm. This waste contains approximately 417 million curies of radioactivity and includes approximately 3 million gallons of sludge and 34 million gallons of salt waste. Additionally, 5 million curies have already been dispositioned by the Defense Waste Processing Facility (DWPF) into glass canisters.

The Accelerated Clean-up Plan (ACP)[1] will result in the acceleration of risk reduction by emptying HLW tanks sooner, shipping HLW glass canisters and Transuranic Waste (TRU) to their geologic repositories sooner and decommissioning the F Canyon facilities and other excess facilities sooner. This document discusses the planned disposition of the material currently stored in the HLW System.

The net result is that by FY 2020, the ACP will have completed disposition and shipment of the canisters to the federal repository, reduced the Life Cycle Cost (LCC) by $7.1 billion and the amount of radioactivity remaining at SRS will be reduced from a planned 188 million curies to less than 20 million curies.

The vast majority of the radioactivity in the HLW System will be incorporated into borosilicate glass via the Defense Waste Processing Facility (DWPF) and will be disposed in a geologic repository. A much smaller amount will be safely disposed in the Saltstone Disposal Facility (SDF), a near surface landfill on the Savannah River Site. SRS has developed a three-pronged strategy to safely and economically treat and dispose of the salt waste. Salt that contains relatively little radioactivity (i.e., low curie salt) will be disposed of at the SDF after removal of interstitial liquid. Salt that contains small amounts of cesium but significantly higher levels of actinides will be processed through the Actinide Removal Process prior to disposal. Salt waste that contains higher levels of actinides and cesium will be processed by the planned Salt Waste Processing Facility (SWPF) to remove both the actinides and the cesium prior to disposal.

Previous plans processed all of the salt waste by treatment in the Salt Waste Processing Facility. This baseline approach does not begin salt waste disposition until FY 2010 and would not complete processing until FY 2027.

The Saltstone Production and Disposal Facilities are engineered to the latest safety and environmental standards. Wastewater contaminants are converted into insoluble species or incorporated into the cement matrix of the saltstone. The final waste form is resistant to leaching of contaminants present in the porous matrix such that the concentrations of radioactive contaminants in the groundwater do not exceed drinking water standards specified in 10CFR 141. Saltstone is classified as a non-hazardous solid waste as defined by US Environmental Protection Agency (USEPA) protocol. The concentration of radionuclides in the solid waste is less than the NRC Class C limits for Low Level Waste (LLW) disposal.

The SRS strategy for cleanup of the HLW System strikes an appropriate balance between cost, schedule, and the protection of the health and safety of the public and environment by directing most of the radioactivity to borosilicate glass, and ultimately a geologic repository. A maximum of 20 million curies, primarily the short-lived Cs-137 will be disposed in the Saltstone Disposal Facility. This represents less than 5% of the 417 million curie HLW inventory.

[1] WSRC-RP-2002-00245, SRS Environmental Management Program Performance Management Plan

Background

The High-Level Waste (HLW) System is a set of six different interconnected facilities. These facilities function as one large treatment plant that receives, stores, and treats wastes from various generators at the Savannah River Site (SRS) and converts them into forms suitable for final disposal. The three major forms are borosilicate glass, which will eventually be disposed of in a federal repository, saltstone, and treated water effluent that is released to the environment.

HLW is a complex mixture of radionuclides, soluble salts, and insoluble sludges. Many of these components are hazardous to human health (e.g. Pu, Cd, Cr, and Hg) or can be serious hazards if not properly managed. All of the wastes are prohibited from continued storage under the Environmental Protection Agency (EPA) regulations. The composition of each of the major effluent streams – glass, saltstone, and decontaminated effluent – is regulated by outside regulatory agencies.

Processing and disposition of these wastes in a safe, cost effective, and environmentally sound manner is critical to accomplishing the HLW system's mission[2]. The 37 million gallons of waste is currently stored in 49 of the HLW system's 51 tanks. Two of the tanks have been isolated, closed, and grouted (Tanks 17 and 20). Twelve of SRS's 51 waste tanks have a leakage history. Twenty-four of the tanks are considered "non-compliant" tanks and do not meet current requirements for secondary containment and leak detection.

High-Level Waste Treatment

Attachment 1 contains a description of the HLW System. Figure 1 schematically illustrates the flow of routine wastes through the HLW system. The numbered streams identified in italics are the interface streams between the various processes.

Sludge Processing

Upon receipt from the canyons, insoluble sludge settles to the bottom of the tank. This insoluble sludge is re-suspended, washed with water to reduce its soluble salt content and sent to DWPF.

The washed sludge is chemically adjusted in the DWPF to prepare the sludge for feed to the glass melter. The highly radioactive waste fraction from Salt Processing (see below) is added to the sludge. Glass frit is added to the mixture and sent to the glass melter. The glass melter drives off the water and melts the wastes into a borosilicate glass matrix, which is poured into a canister. The canistered glass wasteform is sent to an on-site interim storage, and will eventually be disposed in a Federal Repository.

The DWPF began radioactive operations in 1996 and has processed through January, 2003 about 500,000 gallons of sludge waste, including 5 million total curies (as shown in Table 2) and produced 1337 canisters of a projected 5,000 canisters. (The first feed to the DWPF was taken from some of the non-compliant tanks which had a lower curie content per gallon than the waste to be processed later.)

Salt Processing

Saltcake which results from the evaporation of supernate may be considered to be composed of four parts: salt crystals, liquid solution, gas, and insoluble solids. When salt crystals form and accumulate in a waste tank, the interstitial space between the crystals will contain supernate and potentially gas. Gas can consist of trapped air, water vapor, and radiolytically produced gases. Insoluble solids may consist of entrained high-level waste sludge solids or, potentially, insoluble materials that form during concentration in the evaporator.

Cesium solubility was estimated for SRS waste showing that a saturated salt solution would contain 10,738 Ci/gallon.[3] Comparison with the measured radioactivity values for SRS salt solutions indicates the Cs

[2] HLW-2002-00025, "Savannah River High-Level Waste System Plan," Revision 13, March 2002.
[3] WSRC-TR-99-00358, "Hydrological Methods can Separate Cesium from Nuclear Waste Saltcake," J. N. Brooke, K. Stahlie, and J. F. Peters, July 1999.

levels are only 0.2 to 0.9 percent of the saturation level.[3] This low percentage of saturation indicates that most, if not all, of the cesium is in solution and resides in the interstitial liquid with very little cesium being contained in the salt cake. Thus, removing the interstitial liquid from the saltcake matrix will remove almost all of the cesium.

Salt waste is treated by three different methods depending on the radionuclide concentration in the waste. Two of the treatment processes, Low Curie Salt (LCS) and Actinide Removal Process (ARP), begin by draining interstitial liquid from the saltcake to reduce the Cs content of the saltcake. The interstitial liquid is sent back to storage and evaporation. The drained saltcake is then dissolved using water or LLW liquids (potentially DWPF recycle). Any remaining saltcake not treated by LCS or ARP, the concentrated supernate and interstitial liquids removed from the salt cake, will be combined and treated in the Salt Waste Processing Facility (SWPF) to be built in the future.

In LCS, dissolved saltcake is sent to Tank 50 for interim storage until it is sampled, verified to meet the Saltstone waste acceptance requirements and fed to Saltstone.

In ARP, dissolved saltcake low in Cs content is processed through the 512-S facility to remove actinides and strontium which are subsequently concentrated. The process produces decontaminated salt solution and a concentrated MST/sludge stream, containing most of the actinides which is washed with water to remove soluble salts and sent to DWPF for Vitrification. The decontaminated salt solution from this process is sampled and sent to Tank 50 for interim storage until it is transferred to Saltstone.

The remaining salt waste will be treated in the SWPF for the removal of Sr and Actinides by Monosodium Titanate, and treated to remove Cesium by caustic side solvent extraction (CSSX). The washed Sr and actinide sludge slurry is sent to the DWPF for Vitrification. The Cs product stream from CSSX is also sent to the DWPF where it is combined with washed sludge and vitrified. The decontaminated salt solution from this process is sent to either Tank 50 or directly to the Saltstone Facility for disposal.

Current Waste Characteristics

Since the early 1950's SRS has received over 100 million gallons of waste into the HLW system. This waste was neutralized, insoluble sludge settled, and supernate was evaporated to form saltcake such that as of January 2003 only 37 million gallons of waste is stored in F and H tank farm. This waste inventory is shown in Table 1. The radionuclide inventory of the high level waste system is shown in Table 2 along with the radionuclide inventory of glass canisters produced to date.

Table 1
Volume of Waste Currently in Inventory (million gallons)

Sludge Inventory	3
Saltcake Inventory	16
Supernate Inventory	18
Total	**37**

Table 2
Radionuclide Inventory (Curies) at the End of FY02

Isotope	Sludge	Combined Saltcake and Supernate	Current Tank Farm Waste Inventory	Glass Canisters Produced To Date
Cs/Ba-137	11,800,000	201,000,000	**212,800,000**	230,000
Sr/Y-90	189,700,000	5,000,000	**194,700,000**	4,600,000
Tc-99	28,000	20,000	**48,000**	170
I-129	<1	15	**15**	0.06
Actinides	3,800,000	226,000	**4,026,000**	100,000
Other	4,800,000	730,000	**5,530,000**	70,000
Total	**210,128,000**	**206,976,000**	**417,104,000**	**5,000,000**

Final Waste Forms

The vast majority of the radioactivity in SRS's HLW system will be incorporated into borosilicate glass to be disposed in a geologic repository. A much smaller amount (appropriate for management and disposal as LLW) will be disposed in Saltstone, a near surface landfill on the Savannah River Site (SRS). Some residual radioactivity will remain in each Tank Farm and DWPF when these facilities are closed.

The Defense Waste Processing Facility (DWPF) is currently being operated to convert the insoluble sludge portion of HLW into a borosilicate glass. This glass is poured into stainless steel canisters, sealed and stored temporarily at SRS until shipments can begin to a permanent geologic repository. Borosilicate glass has been established as the Best Available Technology (BAT) for the disposal of HLW.

It has been estimated that to incorporate all wastes in the HLW system into borosilicate glass would require the production of 120,000 DWPF canisters, require the production capacity of at least five equivalent DWPFs[4] and take 50 to 100 years to process the waste. Since the highest risk to the environs of SRS is continued HLW storage this approach has been judged unacceptable. Likewise it is not possible to treat HLW and remove 100% of the radioactivity. Therefore, a balance must be struck between the amount of radioactivity disposed in the Saltstone Disposal Facility and the cost and time required to incorporate more radioactivity into the borosilicate glass.

Paramount in striking this balance is the impact on the public and SRS worker. The Saltstone Production and Disposal Facilities are engineered to the latest safety and environmental standards. Saltstone is a solid waste form that is the product of chemical reactions between a salt solution and a blend of cementitious materials (slag, flyash and cement). Several wastewater contaminants are converted into insoluble species or incorporated into the cement matrix. The final waste form is resistant to leaching of contaminants present in the porous matrix and is classified as a non-hazardous solid waste as defined by the United States Environmental Protection Agency (USEPA) protocol[5]. In addition to the wasteform characteristics, the saltstone is cured inside concrete vaults. The vaults, which contain either 6 or 12 cells of dimensions 100 ft by 100 ft by 25 ft high, are constructed of reinforced concrete walls 18 inches thick to provide both a form for the saltstone solidification and provide shielding for the area workers. The vault roof and closure features minimize the amount of water infiltration, and therefore provides protection for the groundwater.

Planned Disposal

The planned disposal for HLW is as follows:

- Insoluble sludge is converted to borosilicate glass,
- Salt waste is split into the three disposition categories - LCS, ARP and SWPF - depending on the initial radioactivity content. The initial radioactive content and initial disposition of each tank has been documented in reference 6.

Since much of the salt waste is in a crystallized saltcake form, and it must be dissolved to be retrieved, the 34 million gallons of salt waste volume expands to approximately 84 million gallons at the time of retrieval. After processing by LCS, ARP and SWPF this volume is over 100 million gallons and due to the dry materials added at saltstone the volume is over 185 million gallons at disposal. Table 3 contains the forecast amount of salt waste for each program segment, including the amount of saltcake and concentrated supernate processed, the equivalent amount of feed to each process at the time of retrieval (at 6.44 M Na concentration), the equivalent amount of solution processed at saltstone, and the amount of saltstone disposed.

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[4] WSRC-RP-98-00165rev 1 High Level Waste Salt Disposition Systems Engineering Team HLW Salt Disposition Alternatives Identification Preconceptual Phase II Summary Report

[5] WSRC-TR-2002-00236, "Saltstone Landfill Design Equivalency Demonstration" August 30, 2002 (Part of the "Application for Revision Z Area Industrial Solid Waste Landfill Permit #025500-1603, submitted to SC DHEC September 2002)

[6] HLW-SDT-2002-00004, "Low Curie Tank Selection Process", January 15,2002

Table 3
Forecast Salt Processing Volumes (kgal) for LCS, ARP, and SWPF (ref.[7])

	Salt Cake Inventory	Concentrated Supernate Inventory	Process Feed at Retrieval (6.44M Na)	Saltstone Feed Stream	Saltstone Disposed
Salt Solution processed via Low Curie Processing	**6,670**		**28,403**	**33,830**	**62,300**
Salt Solution processed via Actinide Removal Processing	**6,300**		**27,769**	**33,000**	**60,800**
Salt Solution processed via Salt Waste Processing Facility	**2,630**	**18,000**	**28,489**	**34,007**	**62,600**
Total Volume (kgal)	**15,600**	**18,000**	**84,662**	**100,837**	**185,700**

Table 4 contains the forecast amount of radioactivity for glass and saltstone at the completion of HLW disposition (FY20) as well as the forecast amount remaining in the tank farm and DWPF at the time of closure. Radioactive decay been excluded for clarity.

Table 4
Radionuclide Inventory- Not Decayed (Curies) at the End of FY2020

Isotope	Saltstone	Total Glass Canisters [a]	Tank Farm Closure	DWPF Closure	Total Waste Inventory	% in Glass
Cs/Ba-137	17,600,000	195,404,200	20,000	5,800	213,030,000	91.7
Sr/Y-90	1,678,000	197,181,800	330,000	110,200	199,300,000	98.9
Tc-99	20,000	28,000	50	5	48,000	58.4
I-129	15	<1	0	0	15	6.0
Actinides	98,000	4,014,000	11,000	3,000	4,126,000	97.3
Other	603,000	4,857,000	50,000	90,000	5,600,000	86.7
Total	**20,000,000**	**401,484,000**	**411,000**	**209,000**	**422,104,000**	**95.1**

a. Includes Radioactivity contained in Canisters already produced

The distribution of the Saltstone actinide radioactivity at disposal is provided in Table 5. The majority (over 80%) of the radioactivity is present as isotopes with half lives of less than 100 years.

Table 5
Actinide Distribution for Waste Disposed at Saltstone

Isotope	Half-Life (yrs)	Percent of Actinide Curies
Pu-238	88	49.8
Pu-241	14	26.0
Am-241	432	16.1
Cm-244	18	5.9
Pu-239	24,110	1.6
Pu-240	7	0.5
Other *	N/A	0.1
Total	**N/A**	**100.0**

* Less than 0.02% of Total Actinide curies each

The programmatic source of the radioactivity disposed in Saltstone is shown in Table 6. Radioactive decay has been excluded for clarity.

[7] HLW-2002-00161, "PMP Supplement to HLW System Plan Rev. 13", December 18, 2002.

Table 6
Programmatic Source for Saltstone Disposal (No Radioactive Decay) FY-2020

Salt Disposition Element	Cs/Ba-137	Sr/Y-90	Tc-99	I-129	Actinides	Other	Total
Low Curie Salt	10,000,000	1,540,500	2,000	2	32,900	603,000	**12,178,400**
Actinide Removal	7,595,000	135,000	2,000	2	32,100		**7,764,100**
Salt Waste Processing Facility	5,000	2,500	16,000	11	33,000		**56,500**
Total	17,600,000	1,678,000	20,000	15	98,000	603,000	**20,000,000**

The Department of Energy (DOE) has designated the SRS as a long term asset stating that "SRS boundaries shall remain unchanged and the land shall remain under the ownership of the federal government" in the SRS Long Range Comprehensive Plan,[8] signifying their intent to maintain control of the SRS in perpetuity. As such, access to the SDF will be restricted and the facility will remain under active management during the time of radioactive decay. Table 7 reflects the radioactivity following 300 years (from today) of decay.

Table 7
Radionuclide Inventory (Curies) FY 2300 (Decayed)

Isotope	Saltstone	Glass Canisters	Total Waste
Cs/Ba-137	17,200	190,800	208,000
Sr/Y-90	1,600	192,600	194,200
Tc-99	20,000	28,000	48,000
I-129	15	<1	15
Actinides	28,000	1,153,800	1,181,800
Other	1,200	10,000	11,200
Total	**68,000**	**1,575,200**	**1,643,200**

Therefore, even though initially, the salt processing initiatives would leave 20 million curies in Saltstone, by 2300 this material would have decayed to less than 100,000 curies.

Regulatory Discussion

HLW is defined in several Code of Federal Regulations (see Attachment 2) based upon the source or location of its generation. The definition as provided in DOE Order 435.1 is:

"High-level waste is the highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and other highly radioactive material that is determined, consistent with existing law, to require permanent isolation".

The term in the definition "sufficient concentrations" has not been developed further. It is therefore not possible to determine by numerical comparison if treatment of HLW results in sufficient decontamination of the waste that it can be treated as another type of waste. The process for determining if sufficient decontamination has occurred is defined in DOE Order 435.1 as the Waste Incidental to Re-Processing (WIR) process.

The WIR process required by DOE Order 435.1 has been compared to the process endorsed by the NRC (see Attachment 3). The processes of the two agencies are similar. The WIR process requires three attributes to be successfully documented:

- The waste must be processed to remove key radionuclides to the maximum extent technically and economically practical

[8] "SRS Long Range Comprehensive Plan" December 2000

- Disposal of the waste must meet Safety requirements comparable to Performance Objectives outlined by the NRC in 10 CFR 61
- The waste must be incorporated into a solid physical form that does not exceed NRC Class C concentration limits.

The Low Curie Salt Alternative has been subjected to the WIR process[9] and DOE[10] has approved the determination that the waste is appropriate for management as LLW and for disposal in the SDF. A WIR evaluation for the ARP is being prepared.

Criterion 1 – Technically and Economically Practical

The Salt Waste Processing Facility (SWPF) will be designed to remove Strontium, Plutonium, Neptunium, limited Uranium, and the vast majority of the Cesium from salt waste. The overall extent of removal is estimated (based on laboratory scale testing) as 99.9988% of the radioactivity. While the planned disposition based on all three salt waste treatment technologies will remove 95.1 % of the radioactivity, a difference of 4.8%. As shown in Tables 5 and 6 over 96% of the radioactivity disposed in Saltstone has a half-life of 30 years or less.

The cost of processing a gallon of saltcake through the proposed SWPF is estimated at $104.06 while the cost of Low Curie Salt Processing is estimated at $17.48, a difference of $86.58 per gallon. In other words to remove the remaining 4.8% of the radioactivity would require 6 times the investment of funds and would delay the risk due to the delay resulting from the construction of the full scale Salt Waste Processing Facility.

Criterion 2- Performance Objectives

The Saltstone Disposal Facility is a modern disposal facility designed to protect the environment, the public and the worker. DOE Order 435.1 specifies the following Performance Objectives for the SDF:

- Dose to the public shall not exceed 25mrem/yr total EDE from all pathways
- Dose to the public via the air pathway shall not exceed 10 mrem/yr total EDE
- Release of Radon shall be less than an average flux of 20pCi/m^2/sec at the surface of the disposal facility
- Dose to hypothetical persons assumed to intrude on the SDF shall not exceed 100 mrem/yr for chronic exposures or 500 mrem/yr for acute exposures

In addition, since the SRS is a closure site under CERLA, DOE has established a more restrictive performance objective to protect groundwater,

- Concentrations of radioactive contaminants in the groundwater shall not exceed drinking waster standards specified in 10CFR 141, generally 4 mrem/yr EDE

These Performance Objectives are essentially identical to the Performance Objectives required by the NRC[11] (see also Attachment 2) as specified in 10CFR61. Compliance with these standards is documented in reference[12], which documents the maximum inventory (Ci/vault) for each radionuclide of interest. Table 8A and B compare the limits for groundwater protection and hypothetical intruder to the planned disposal.

Tables 8A and 8B clearly indicate the planned waste disposition meets the performance objective for disposal in the SDF.

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[9] HLW-SDT-2002-00281 rev. 1, "Waste Incidental to Reprocessing Evaluation for Disposing Saltcake to Saltstone", February, 2002.
[10] Letter Anderson to Piccolo, "Resubmittal of Waste Incidental to Reprocessing (WIR) Document (Your Letter 2/14/02)", March 8, 2002.
[11] WSRC-RP-2001-00341 "Comparison of LLW Disposal Performance Objectives 10 CFR 61 and DOE Order 435.1", March 2001
[12] WSRC-TR-2002-00456, "Special Analysis: Reevaluation of the Inadvertent Intruder, Groundwater, Air, and Radon Analysis for the Saltstone Disposal Facility", October 23, 2002.

Criterion 3- NRC Class C

The Saltstone product is a solid and as shown in Table 9 the concentration of radionuclides in the solid waste is less than the NRC Class C limits.

A similar WIR evaluation for the decontaminated salt solution produced by the ARP is currently under development and is expected to show that material treated by the ARP can be managed as LLW similar to the LCS material.

Table 8A
SDF Disposal Limits for Ground Water Protection vs. Planned Disposition

Nuclide	Half-Life	Limiting Pathway	Maximum Landfill Inventory (ci)	Planned Disposition	Fraction of Allowable for Each Pathway *a*
Se-79	1,100,000 y	Groundwater	3,930	200	0.050
Tc-99	211,000 y	Groundwater	398,000	20,000	0.001
I-129	15,700,000 y	Groundwater	20	15	0.750
Np-237*b*	2,144,000 y	Groundwater	102,000	2,000	0.020
					Total Sum of Fraction 0.821

a Sum of the Fractions for Each Pathway must not exceed 1.0
b Actual disposals will be limited by the definition of TRU waste of 100 nCi/gram

Facility Worker Safety

The two most important parameters in Safety analysis are the dose received during accident conditions and during routine operation of the facility. An Auditable Safety Analysis (ASA) is documented in reference [13] to demonstrate the safe operation to the facility worker.

The ASA documents a "worst case" accident dose to the facility worker is a maximum 10 mrem EDE. This maximum dose puts the facility in a "Radiological Facility" category, the lowest category recognized by the DOE.

The routine facility radiation fields are the highest in the Saltstone Production Facility (SPF) process room. This room is "controlled access", entry is only allowed after processing has completed, piping and equipment has been flushed and radiation surveys indicate it is safe to enter. The control room has a radiation field that is acceptable for continuously occupied spaces of less than 0.5mR/hr.

Radiation fields in the SDF during production are estimated as 10mR/hr along the side of the vault and 100mR/hr along the top of the vault. Both of these areas are "controlled access". Entry into these areas is required only for maintenance activities.

At closure of the SDF when the backfill, clay cap and overburden have been installed around the vaults the radiation levels decrease to approximately 300,000 times less than background radiation.

[13] WSRC-TR-2002-00417, "Auditable Safety Analysis for the Saltstone Facility", October 2002.

Table 8B
SDF Disposal Limits for Hypothetical Intruder Protection vs. Planned Disposition

Nuclide	Half-Life	Limiting Pathway	Maximum Landfill Inventory (Ci)	Planned Disposition *d*	Fraction of Allowable for Each Pathway *a*
Al-26	717,000 y	Intruder	2,340		
Co-60	5.3 y	Intruder	4.3×10^{12}		
Sr-90	28.8 y	none *c*	none	1,663,000	0
Nb-94	20,300 y	Intruder	12,410		
Sn-126	100,000 y	Intruder	14,600	800	0.054
Cs-137	30 y	Intruder	1.4×10^{10}	17,500,000	0.001
Eu-152	13.5 y	Intruder	4.7×10^{9}		
Eu-154	8.6 y	Intruder	1.1×10^{11}		
Ra-226	1,600 y	Intruder	7,080		
Ra-228	5.7 y	Intruder	1.3×10^{11}		
Ac-227	21.7 y	Intruder	1.7×10^{11}		
Th-229	7,340 y	Intruder	130,000		
Th-230	75,400 y	Intruder	4,900		
Th-232	1.4×10^{10} y	Intruder	2,340		
Pa-231	32,760 y	Intruder	290,000		
U-232	68.9 y	Intruder	2.2×10^{6}		
U-233	159,200 y	Intruder	204,000		
U-235	70,380,000 y	Intruder	1.3×10^{6}		
U-238	4.4×10^{10} y	Intruder	800,000		
Pu-238*b*	87.7 y	Intruder	189,800	48,804	0.257
Pu-239*b*	24,110 y	Intruder	2.6×10^{11}	1,570	0
Pu-241 *b*	14.3 y	Intruder	1.2×10^{11}	25,480	0
Pu-244*b*	80,000,000 y	Intruder	57,670		
Am-241*b*	432 y	Intruder	3.6×10^{9}	15,788	0
Am-242m*b*	141 y	Intruder	1.5×10^{8}		
Am-243 *b*	7,370 y	Intruder	3.9×10^{6}		
Cm-242 *b*	163 d	Intruder	3.8×10^{10}		
Cm-243 *b*	28.5 y	Intruder	1.7×10^{14}		
Cm-245 *b*	8,500 y	Intruder	9.5×10^{7}		
Cm-247 *b*	15,600,000 y	Intruder	2.7×10^{5}		
Cm-248 *b*	348,000 y	Intruder	6.6×10^{8}		
Bk-249 *b*	330 d	Intruder	8.8×10^{9}		
Cf-249 *b*	351 y	Intruder	2.2×10^{6}		
Cf-251 *b*	13.1 y	Intruder	2.9×10^{7}		
Cf-252 *b*	2.6 y	Intruder	8.7×10^{13}		
					Total Sum of Fraction 0.312

a Sum of the Fractions for Each Pathway must not exceed 1.0
b Actual disposals will be limited by the definition of TRU waste of 100 nCi/gram or 197,100 Ci at the landfill capacity or 98,000 Ci at the planned disposal quantity.
c Disposal of Sr-90 is not limited by the Hypothetical Intruder but it is included here since it is a significant fraction of the planned disposal curies
d Only radionuclides that are a significant fraction of the planned disposal is provided, other radionuclides will be verified within limits prior to disposal.

Table 9
Comparison of NRC Class C Limits vs. Expected Concentrations in the SDF

Radionuclide	Low Curie Saltstone (nCi/g)	NRC Class C Limits (nCi/g)
C-14	0.004	4,710
Cs-137	47,000	2,710,000
Ni-63	0.009	412,000
Sr-90	3,600	4,120,000
Tc-99	28	47,100
Total Alpha Emitters	99	100

Comparison to Barnwell and the Current SRS LLW Disposal Sites

The planned disposition at Saltstone can be compared to the commercial LLW disposal sites at Barnwell as well as another DOE LLW disposal site on the SRS. Table 10 compares the disposals at Barnwell from January, 1986 to September, 2002[14] to the planned disposal for Saltstone.

Table 10
Comparison of Planned Saltstone Disposal to Commercial LLW Disposal Sites

Disposal Facility	Total Curies
Barnwell	7,197,032
Saltstone	20,000,000

It should be noted that Barnwell began operations in 1972 and that data prior to 1986 is not available via the internet. It is estimated that 3,000,000 curies was disposed at Barnwell between 1972 and 1986 bringing the total to 10 million curies or about half the planned Saltstone disposal. Since over 96% of the planned disposal at Saltstone has a half-life of 30 years or less an isotopic comparison to the other disposal facilitates is not judged to be necessary.

Table 11 compares the planed Saltstone disposal to the SRS E-Area Disposal Facility.

Table 11
Comparison of Planned Saltstone Disposal to E-Area

Disposal Facility	Total Curies
E-Area Navel Fuels Pads[15]	2,560,000
E-Area Low Active Waste[15]	1,660,000
E- Area Intermediate Level Waste[15]	920,000
E-Area Melt and Dilute Planned Disposal	20,000,000
Total E-Area	**24,140,000**
Saltstone	**20,000,000**

[14] Manifest Information Management System, http://mims.mactec.com/#
[15] WSRC-RP-97-311, Composite Analysis of E-Ears Vaults and Saltstone Disposal Facilities, September, 1997.

The E-Area Vault planned (life cycle) disposal is taken from reference 15 except for the contribution for the Melt and Dilute Process which has been cancelled. The Melt and Dilute Process for the disposition of excess plutonium included off gas scrubbers which adsorbed Cs-137 and would have been disposed at the E-Area vaults. The amount of Cs-137 to be disposed was estimated at 20 million curies.

Comparison to Other Baselines

Waste Disposition at the SRS has evolved over time. Most recently emphasis has been to ship waste from the SRS to the national repositories - the Waste Isolation Pilot Plant (WIPP) for transuranic waste and the geologic repository for HLW at Yucca Mountain - sooner. The disposition of salt waste has been modified (as discussed extensively above) to treat the waste in advance of the construction and start-up of the SWPF. This action results in more curies being disposed in Saltstone. While at the same time, the plan for disposition of excess radioactive materials (e.g. plutonium) has changed resulting in a deceased curies being disposed in the E-Area Vaults of the SRS. The decrease in E-Area vault disposal is about 50% of the increase in the Saltstone. Figure 1 shows the DOE Environmental Management (EM) curies remaining at SRS for the baseline and current planned disposition (labeled "Accelerated Clean-up") alternatives.

Figure 1
Radioactivity Remaining at SRS Originating from All Environmental Management Activities

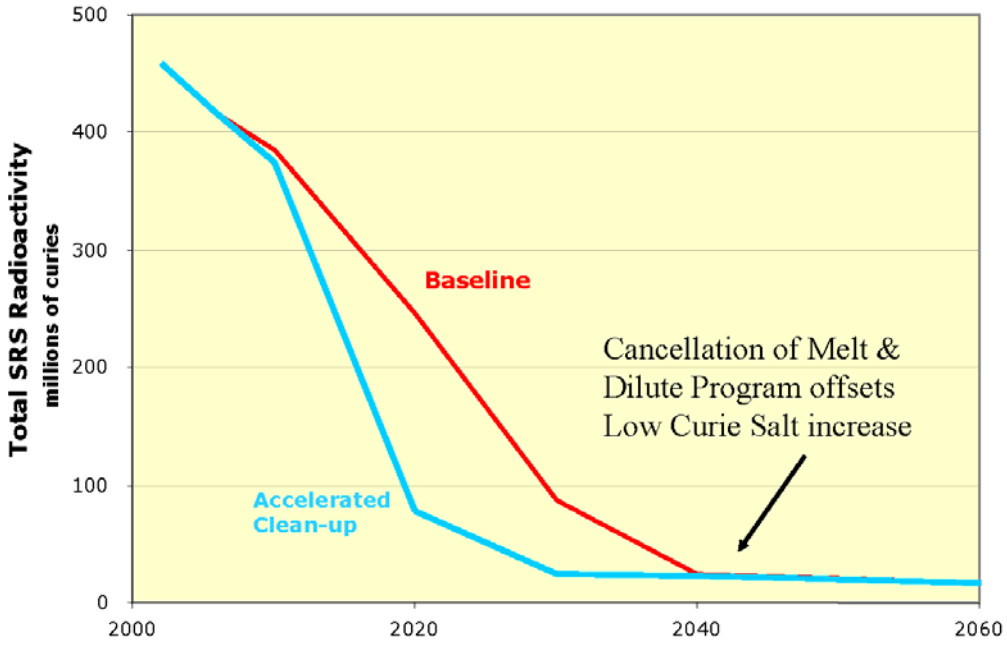
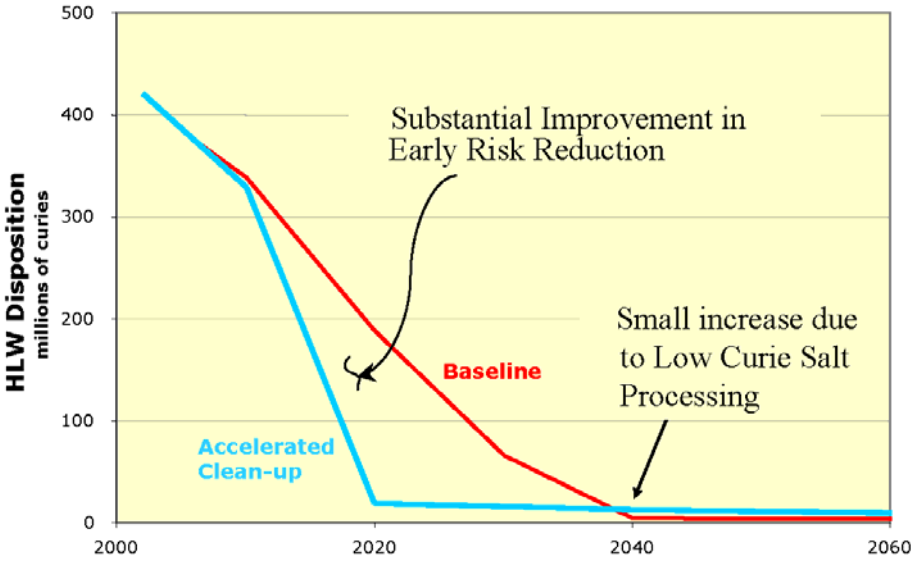


Figure 2 shows the EM curies remaining at SRS which originate in the HLW system. This figure shows the increase in the curies disposed in Saltstone (about 8 million curies in CY 2040). Figure 2 also shows that these curies decrease rapidly. The curies disposed in saltstone are predominately Cs-137, a radionuclide with a half-life of only 30 years. By CY 2100 less than 1% of the existing curies remain at SRS.

Figure 2
Radioactivity Remaining at SRS Originating in the HLW System



Disposal of HLW also includes the residual activity left in the F and H Tank Farms and in the DWPF upon closure of these facilities. Table 12 summarizes the prior baseline including closure activities while Table 13 summarizes the planned baseline including closure activities without radioactive decay being taken into consideration. It would indicate that up to 30 times as much radioactivity would be left at SRS. However, when radioactive decay is taken into consideration for 300 years, as shown in Tables 14 and 15, only 2.6 times as much radioactivity remains on SRS.

Table 12
Prior HLW System Baseline Summary (Not decayed)

Isotope	Prior Saltstone Baseline	Tank Farm Closure	DWPF Closure	Prior HLW System Baseline
Cs/Ba-137	5,000	20,000	5,800	30,800
Sr/Y-90	20,000	330,000	110,200	460,200
Tc-99	20,000	50	5	20,055
I-129	15	0	0	15
Actinides	9,700	11,000	3,000	23,700
Other	0	50,000	90,000	140,000
Total	**54,700**	**411,000**	**209,000**	**674,700**

Table 13
HLW System Radioactivity Remaining at SRS with Accelerated Clean-up (Not Decayed)

Isotope	Planned Saltstone Disposal	Tank Farm Closure	DWPF Closure	HLW System Radioactivity at SRS
Cs/Ba-137	17,600,000	20,000	5,800	17,625,800
Sr/Y-90	1,678,000	330,000	110,200	2,118,200
Tc-99	20,000	50	5	20,055
I-129	15	0	0	15
Actinides	98,000	11,000	3,000	112,000
Other	603,000	50,000	90,000	743,000
Total	**20,000,000**	**411,000**	**209,000**	**20,619,000**

Table 14
Prior HLW System Baseline Summary FY2300 (Decayed)

Isotope	Prior Saltstone Baseline	Tank Farm Closure	DWPF Closure	Prior HLW System Baseline
Cs/Ba-137	5	20	6	31
Sr/Y-90	20	300	110	430
Tc-99	20,000	50	5	20,055
I-129	15	0	0	15
Actinides	2800	3200	900	6,900
Other	0	100	200	300
Total	**22,800**	**3,700**	**1,220**	**28,000**

Table 15
HLW System Radioactivity Remaining at SRS with Accelerated Clean-up FY2300 (Decayed)

Isotope	Planned Saltstone Disposal	Tank Farm Closure	DWPF Closure	HLW System Radioactivity at SRS
Cs/Ba-137	17,200	20	6	17,226
Sr/Y-90	1,600	300	110	2,010
Tc-99	20,000	50	5	20,055
I-129	15	0	0	15
Actinides	28,000	3200	900	32,100
Other	1,200	100	200	1,500
Total	**68,000**	**3,700**	**1,220**	**73,000**

Alternative Salt Waste Disposition

In the Salt Processing Alternatives Final Supplemental Environmental Impact Statement[16] DOE compared four cesium and actinide removal alternatives to a "No Action" alternative. In the No Action alternative, all of the salt waste containing 120 million curies Cs-137 (230 million curies of Cs/Ba-137) including 500 curies of long half-life radionuclides (Tc-99 and Pu-239) was released to the surface waters of the at SRS assuming no loss of contaminants during overland flow. The long term impacts were summarized for various scenarios in Table S-7 of the SEIS, and in Table 16 below. The positive impact of waste disposal as Saltstone is clearly evident by comparison of the No Action Alternative to the current disposition plan The SEIS concluded "The presence of 120 million curies in the vaults from the direct Disposal in Grout alternative would be evident in the long term impacts, but the impacts of all the alternatives would be described as small."

[16] DOE/EIS-0082-S2, "Savannah River Site Salt Processing Alternatives, Final Supplemental Environmental Impact Statement", June 2001

Table 16
Summary Comparison of Long-Term Impacts by Salt Processing Alternative

Parameter	No Action	Planned Salt Waste Disposal
Nitrate concentration at 100-meter well (mg/L)[a]	NA	16[c]
Radiation dose (millirem per year) from 100-meter well	640[b]	0.42
Radiation dose from Agricultural Scenario (millirem per year)	NA	NA – Limited by Residential Scenario
Radiation dose from Residential Scenario at 100 years postclosure (millirem per year)	2,320,000[d]	100[e]

a. Nitrate MCL is 10 mg/L.
b. Based on consumption of contaminated surface water at Fourmile Branch.
c. Derived from " Application for Revision – Z Area Industrial Solid Waste Permit #025500-1603", September 2002 (WSRC-TR-2002-236)
d. Based on external radiation in the area of the tank farm.
e. The maximum allowable per the SDF Performance Objectives (WSRC-TR-2002-00456, October 23, 2002.
NA = not applicable.

Conclusions

The current SRS plan for disposing of waste in the HLW system maximizes risk reduction by reducing the greatest risk of continued storage of liquid waste as soon as possible. The plan strikes an appropriate balance between cost and schedule, while providing for protection of the health and safety of the public and environment by directing most of the radioactivity to the borosilicate glass, and ultimately a geologic repository. . A maximum of 20 million curies, primarily the short lived Cs-137 will be disposed in the Saltstone Disposal Facility. This represents less than 5% of the current 417 million curie HLW inventory. The SDF is an engineered disposal facility which provides for low water infiltration, low contaminant leaching, and shielding for the SRS worker. The disposal meets all environmental regulations and standards.

High-Level Waste Treatment

Figure 1 schematically illustrates the flow of routine wastes through the HLW system. The numbered streams identified in italics are the interface streams between the various processes.

F and H Tank Farms

Fresh Canyon wastes (Interface Stream 1) are received into HLW Storage and Evaporation which separates into an insoluble sludge and aqueous supernate. The purpose of the Storage and Evaporation process is to safely manage these wastes until downstream processes are available for further processing. The supernate is evaporated to a solid saltcake to reduce the volume and mobility of the waste. The decontaminated overheads from the evaporators are Low Level Waste and sent to the ETF (Interface Stream 2).

The insoluble sludge that settles to the bottom of waste receipt tanks in Storage and Evaporation is re-suspended, washed with water to reduce its soluble salt content and sent to DWPF (Interface Stream 6). Wash water from this process is sent back to Storage and Evaporation.

Salt Processing

Saltcake may be considered to be composed of four parts: salt crystals, liquid solution, gas, and insoluble solids. When salt crystals form and accumulate in a waste tank, the interstitial space between the crystals will hold supernate and potentially gas. Gas can consist of trapped air, water vapor, and radiolytically produced gases. Insoluble solids may consist of entrained high-level waste sludge solids or, potentially, insoluble materials that form during concentration in the evaporator.

Cesium solubility was estimated for SRS waste where a saturated solution would contain 10,738 Ci/gallon.[17] Comparison with the actual radiation values for SRS salt solutions indicates the Cs levels are only 0.2 to 0.9 percent of the saturation level[14]. This indicates that most, if not all, of the cesium is in solution and resides in the interstitial liquids of the salt cake. Thus, removing the interstitial liquid from the saltcake will remove almost all of the cesium.

Salt waste is treated by three different methods depending on the radionuclide concentration in the waste. Two of the treatment processes, Low Curie Salt (LCS) and Actinide Removal Process (ARP), begin by draining interstitial liquid (Interface Stream 13) from the Saltcake to reduce the Cs content to acceptable levels. The interstitial liquid is sent back to Storage and Evaporation. The drained saltcake is then dissolved using water or LLW liquids (potentially DWPF recycle). Any remaining Saltcake not treated by LCS or ARP along with concentrated supernate will be combined (Interface Stream 5) and treated in the Salt Waste Processing Facility (SWPF) to be built in the future.

In LCS, dissolved saltcake meeting the Saltstone requirements is sent to Tank 50 (Interface Stream 3), for analysis and interim storage until it is sampled and fed to Saltstone (Interface Stream 11).

In ARP dissolved saltcake low in Cs content (Interface Stream 4) is processed to remove actinides and strontium, which are concentrated. The process produces a concentrated MST/sludge stream, containing most of the actinides which is washed with water to remove soluble salts and sent to DWPF for Vitrification. The decontaminated salt solution from this process is sent to Tank 50 (Interface Stream 8), for analysis and interim storage until it is sampled and fed to Saltstone (Interface Stream 11).

Dissolved saltcake not treated by LCS or ARP and concentrated supernate are combined (Interface Stream 5) and treated in the SWPF for the removal of Sr and Actinides by Monosodium Titanate, and treated to remove Cesium by solvent extraction. The washed Sr and actinide sludge slurry is sent to the DWPF for Vitrification. The Cs product stream from CSSX (Interface Stream 10) is also sent to the DWPF where it is combined with washed sludge and vitrified. The decontaminated salt solution from this process is sent to either Tank 50 or directly to the Saltstone Facility for disposal.

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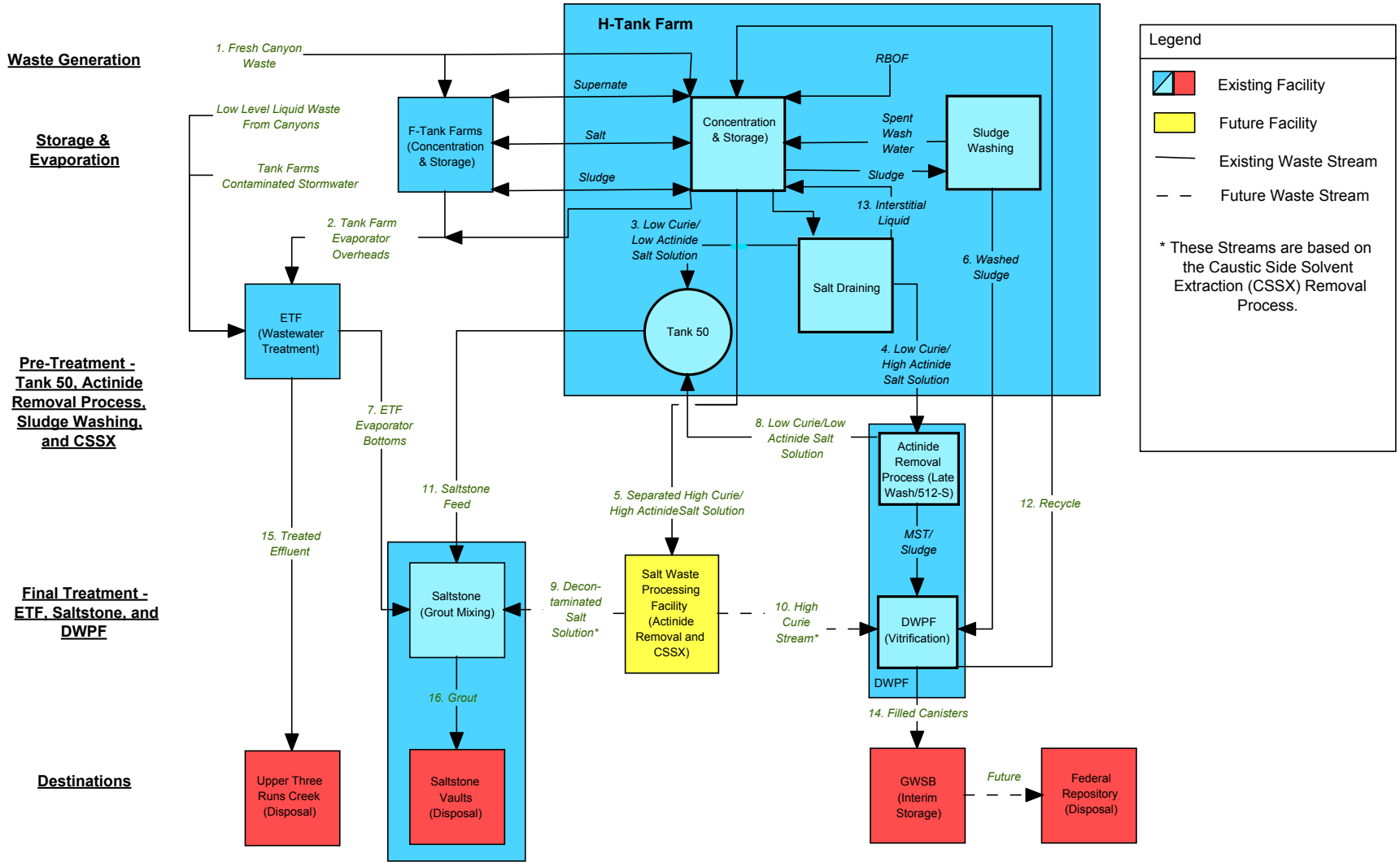
[17] WSRC-TR-99-00358, "Hydrological Methods can Separate Cesium from Nuclear Waste Saltcake," J. N. Brooke, K. Stahlie, and J. F. Peters, July 1999.

Figure 1 - HLW System Facilities and Interfaces



Sludge Processing

The washed sludge (Interface Stream 6) is chemically adjusted in the DWPF to prepare the sludge for feed to the glass melter. The Sr and actinide sludge slurry product and/or Cs product is added to the sludge in DWPF. The mixture is then mixed with glass frit and sent to glass melting. The glass melter drives off the water and melts the wastes into a borosilicate glass matrix, which is poured into a canister. The canistered glass wasteform (Interface Stream 14) is sent to on-site interim storage to await shipment to and disposal in a Federal Repository.

The water vapor driven off from the melter along with other aqueous Low Level Waste streams generated throughout the DWPF Vitrification building are recycled to Storage and Evaporation for evaporation, storage, and eventual further processing (Interface Stream 12).

Wastewater Treatment

The Tank Farm Evaporator Overheads are combined with overheads from evaporators in the F- and H-area Separations processes and other low-level streams from various waste generators. This mixture of low-level wastes is sent to the ETF (Interface Stream 2).

In the ETF, these low-level wastes are decontaminated by a series of cleaning processes. The decontaminated water effluent is sent to the H-area outfall and eventually flows to on-site creeks and the Savannah River (Interface Stream 15). The contaminants removed from the water are concentrated and sent to Tank 50H (Interface Stream 7) to be sent to Saltstone (Interface Stream 7).

Low Level Waste Treatment

Low curie and low curie/low actinide salt solutions (Interface Streams 3 and 8) are combined in Tank 50. This mixture of wastes is sent to Saltstone (Interface Stream 11) along with the decontaminated salt solution from the SWPF (Interface Stream 10) and the ETF concentrate (Interface Stream 7).

The Saltstone Production Facility (SPF) and the Saltstone Disposal Facility (SDF) are both located in Z Area at the SRS. The Saltstone facilities are used to: 1) process aqueous salt solution waste in the SPF to generate a Low-Level Waste wasteform known as Saltstone; and 2) dispose of Saltstone in a safe and environmentally sound manner by placing the Saltstone in concrete vaults that are located in the SDF.

High Level Waste
Definitions

Definitions of High-Level Waste:

The Atomic Energy Commission (AEC) defined high-level waste (HLW) in 10 CFR 50 Appendix F in 1970 as:

" ... those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel. "

The **Nuclear Regulatory Commission** (NRC) defined HLW in 10 CFR 60 as:

"... (1) Irradiated reactor fuel, (2) liquid wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel, and (3) solids into which such liquid wastes have been converted. "

The **Nuclear Waste Policy Act (NWPA) of 1982** defined HLW as:

"... (a) The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (b) Other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation".

The **Department of Energy** (DOE) definition of HLW is similar to the NWPA of 1982 except for the reference to the "Commission" and is as follows:

"High-level waste is the highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and other highly radioactive material that is determined, consistent with existing law, to require permanent isolation".

Waste Incidental to Reprocessing Topic Paper

Question: How similar are the processes used by the DOE and the NRC for determining what is "Incidental Waste"?

Answer: The NRC does not have a process for determining what is incidental waste because commercial reactor fuel is not reprocessed in the United States. However the NRC endorses the concept of incidental waste and has given DOE guidance on determining when a waste may be considered incidental. DOE order 435.1 is based on this guidance and other exchanges of information between the DOE and the NRC, including items published in the Federal Register (FR) and the Code of Federal Regulations (CFR). The order requires that wastes classified as incidental meet low-level waste performance requirements that are comparable to NRC requirements for disposal of low-level waste.

Background and Discussion:

Definition of Incidental Waste:

The concept of incidental waste has a long history at NRC. In the Statement of Proposed Policy (*34 FR 8712*) 6/3/69 for Appendix D, 10 CFR Part 50, "Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities," the Atomic Energy Commission (AEC) noted that the term high-level waste, as used in the proposed Appendix D, did not include all wastes originating from (spent nuclear fuel) reprocessing plant operations (Paragraphs 6 and 7). Such wastes included waste streams such as ion exchange beds, sludges, and contaminated laboratory items, clothing, tools, and equipment. Additionally, this category included radioactive hulls and other irradiated and contaminated fuel structural hardware. Although this language (Paragraphs 6 and 7) concerning incidental waste was *deleted* from the final Policy under Appendix F, pending additional study (*35 FR 17530- 17533*), the principle of incidental wastes has been continually supported by both the Department of Energy and the NRC.

In its Advance Notice of Proposed Rulemaking for the Definition of High-Level Radioactive Waste at 10 CFR Part 60 (*52 FR 5992-6001*), the NRC introduced the term incidental wastes and stated that high-level waste does not include such waste streams. Additionally, the Commission stated (footnote 1, *52 FR 5993*) that "incidental wastes generated in further treatment of HLW (e.g., decontaminated salt with residual activities on the order of 1,500 nCi/g Cs-137, 30 nCi/g Sr-90, 2 nCi/g Pu, as described in the Department of Energy's FEIS on long-term management of defense HLW at the Savannah River Plant, DOE/EIS-0023, 1979) would also, under the same reasoning, be outside the proposed Appendix D definition," if they met certain chemical concentrations. Additionally, in the NRC's Proposed Rule for 10 CFR Part 61, for shallow-land disposal of radioactive waste, the Commission stated that the preferable construction of the statute "...is to conform to the traditional definition (for high-level waste). Under this approach, materials that are HLW for purposes of the licensing-jurisdiction provisions of the *Energy Reorganization Act of 1974* will also be regarded as high-level waste under the *Nuclear Waste Policy Act of 1982*, as amended. This would include the primary reprocessing waste streams at DOE facilities, though not the incidental wastes produced in reprocessing" (*53 FR 17709*).

Waste Incidental to Reprocessing Method as defined in DOE 435.1

DOE order 435.1 provides two processes to determine incidental waste.

Citation: wastes that meet the description included in the proposed 1969 Appendix D. (See first paragraph of the incidental waste definition above)

Evaluation: the evaluation process is taken directly from a response to a petition regarding disposal of waste at the Hanford site, the NRC (States of Washington & Oregon: Denial of Petition for Rulemaking, *58 FR 12342-12347* and also included in letter from Bernero (USNRC) to Lytle (DOE) (transmitted March 2, 1993)) commented that: "Assuming implementation of DOE's plans as described above, the Commission concludes that any radioactive material from the double shell tanks that is deposited in the grout facility would not be high-level radioactive waste subject to NRC's licensing jurisdiction. The responsibility for safely managing those wastes rest with the Department of Energy. The basis for the Commission's

conclusion is that the reprocessing wastes disposed of in the grout facility would be 'incidental' wastes because of DOE's assurance that they:

(1) have been processed (or will be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical;

(2) will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in 10 CFR Part 61; and

(3) are to be managed, pursuant to the *Atomic Energy Act of 1954,* as amended, so that safety requirements comparable to the performance objectives set out in 10 CFR Part 61 are satisfied." (*58 FR 12345*)

Technical Requirements:

Clearly the evaluation process requires conformance to 10CFR Part 61 both in the absolute concentrations of radioactive materials (Class C limits) and in meeting the performance objectives. 10 CFR Part 61 are Low Level Radioactive Waste disposal requirements. The Low Level Waste disposal requirements also contained in DOE order 435.1 have been compared to the 10 CFR Part 61 requirements (WSRC-RP-2001-00341 rev. 0) and determined to be equivalent or more stringent. Thus, any waste determined to be incidental by DOE meets the same standards the NRC applies to commercial low-level waste disposal sites.

Specifically these requirements for LLW disposal are contained in four categories as tabulated below:

Category	**10CFR Part 61 vs DOE 435.1**	**DOE 435.1 Additional**
General Requirements	Both require specific performance objectives to be met as detailed below.	
Protection of the General Population	Both require a measure of protection of 25mrem/year to an individual. The two differ only by dose methodology required. The 10 CFR Part 61 uses the ICRP 2 methodology (the standard for the time Part 61 was written) while the DOE order uses a Total Effective Dose Equivalent (TEDE) methodology currently in use. The NRC currently recommends the use of ICRP-30 dose methodology. Both require a Performance Assessment to provide reasonable assurance of meeting this requirement.	The DOE order requires an assessment of the impact to water resources. This is currently measured against the drinking water standards.
Protection of Individuals from Inadvertent Intrusion	10 CFR Part 61 requires intruder calculations based on 500 mrem per year dose. The DOE requirement is 100 mrem for chronic exposure and 500 mrem for acute exposure. Thus the DOE requirement is more stringent.	
Stability of the Disposal Site after Closure	Both require similar measure to protect the long term stability of the disposal site.	

Although key radionuclides are not defined by the NRC in either the Denial of Petition for Rulemaking or the letter from R. Bernero to J. Lytle, dated March 2, 1993, it is generally understood that key radionuclides applies to those radionuclides that are controlled by concentration limits in 10 CFR 61.55. Specifically these are: long-lived radionuclides, C-14, Ni-59, Nb-94, Tc-99, I-129, Pu-241, Cm-242, and alpha emitting transuranic nuclides with half-lives greater than five years and; short-lived radionuclides, H-3, Co-60, Ni-63, Sr-90, and Cs-137. In addition, key radionuclides are those that are important to satisfying the performance objectives of 10 CFR Part 61, Subpart C. Analysis to date at DOE sites indicates other isotopes important to satisfying these performance objectives include Se-79, Sn-126, and Np-237.

Acceptance and Promulgation of the WIR Methodology

A similar characterization was made for the West Valley Demonstration Project in the Technical Evaluation Report prepared by the NRC Office of Nuclear Material Safety and Safeguards, dated November 1988, which concluded there is reasonable assurance that the cement solidification of the decontaminated supernatant (incidental waste) will meet the waste form stability requirements of 10 CFR Part 61 (NRC Technical Evaluation, 11/88). This is an implicit recognition by the NRC that the separated low-activity fraction of high-level waste need not be managed and disposed as high-level waste.

Also, the NRC was consulted on the closure of Savannah River Tanks 17 and 20. In determining that the closed tanks were not high-level waste, SRS followed the guidance in the letter from Bernero (USNRC) to Lytle (DOE) since it was before the issuance of DOE order 435.1. The NCR staff reviewed the closure plans and performance assessments in great detail and stated no objections to the application of the incidental waste methodology in January 1997.

The NRDC petitioned the NRC on July 28, 1998 to assume and exercise immediate licensing authority over all tanks at SRS because DOE was “decommissioning” HLW tanks and “classifying” waste as incidental waste. The NRC denied this petition on October 2, 2000 stating that the “NRC does not have jurisdiction over defense HLW stored at Savannah River” and that the “NRC has in the past recognized the concept of incidental waste”.

Incidental Waste Determination Authority:

The question of whether the NRC or DOE has the authority to make incidental waste determinations (using the evaluation process) was raised by NRC Commissioner Curtiss in December 1992 (SECY-92-391), as a precursor to the Commission’s action on the 1993 Denial of Petition for Rulemaking. In response, the NRC staff (memo for Commissioner Curtiss from J. M. Taylor, 1/14/93) stated that DOE has the responsibility to make an initial determination, and if DOE concludes that the action is not subject to NRC jurisdiction, then DOE can undertake the activity without involving the NRC in any manner. However, if DOE concludes that NRC jurisdiction is unclear (i.e., the waste may be high-level waste and therefore potentially subject to NRC licensing), then DOE has two options: (1) consult with the NRC and then make a decision based on the results of the consultation; or (2) proceed without communication with the NRC.

The staff response then cites the proposed letter from Bernero (USNRC) to DOE (transmitted March 2, 1993) that the NRC would call upon DOE to provide relevant technical information that would enable the NRC to make its own determination, should that be appropriate. (Although this decision applied to the Hanford case only, DOE’s interpretation, based on discussions with NRC staff, is that it can be applied more broadly through DOE M 435.1-1.) These two memoranda are interpreted to mean that the NRC expects the DOE to consult with them for those waste streams that the DOE has some question of whether the waste stream is high-level waste. In addition, as discussed in the guidance to Section I.2.F.(18), the NRC has licensing authority over DOE facilities “authorized for the express purpose of subsequent long-term storage of high-level radioactive waste generated by DOE and its predecessor agencies”.