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The Savannah River Site Accelerated Clean-Up Mission: Salt Waste Disposal

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Abstract

High Level Waste at SRS is in two principal forms: Sludge and Salt (water-soluble waste). This paper discusses a plan to dispose of the salt waste which will save up to $5.4 billion (life cycle). The flowsheet for each process is described, annual production estimates shown, regulatory issues discussed and the progress to date summarized.

Introduction

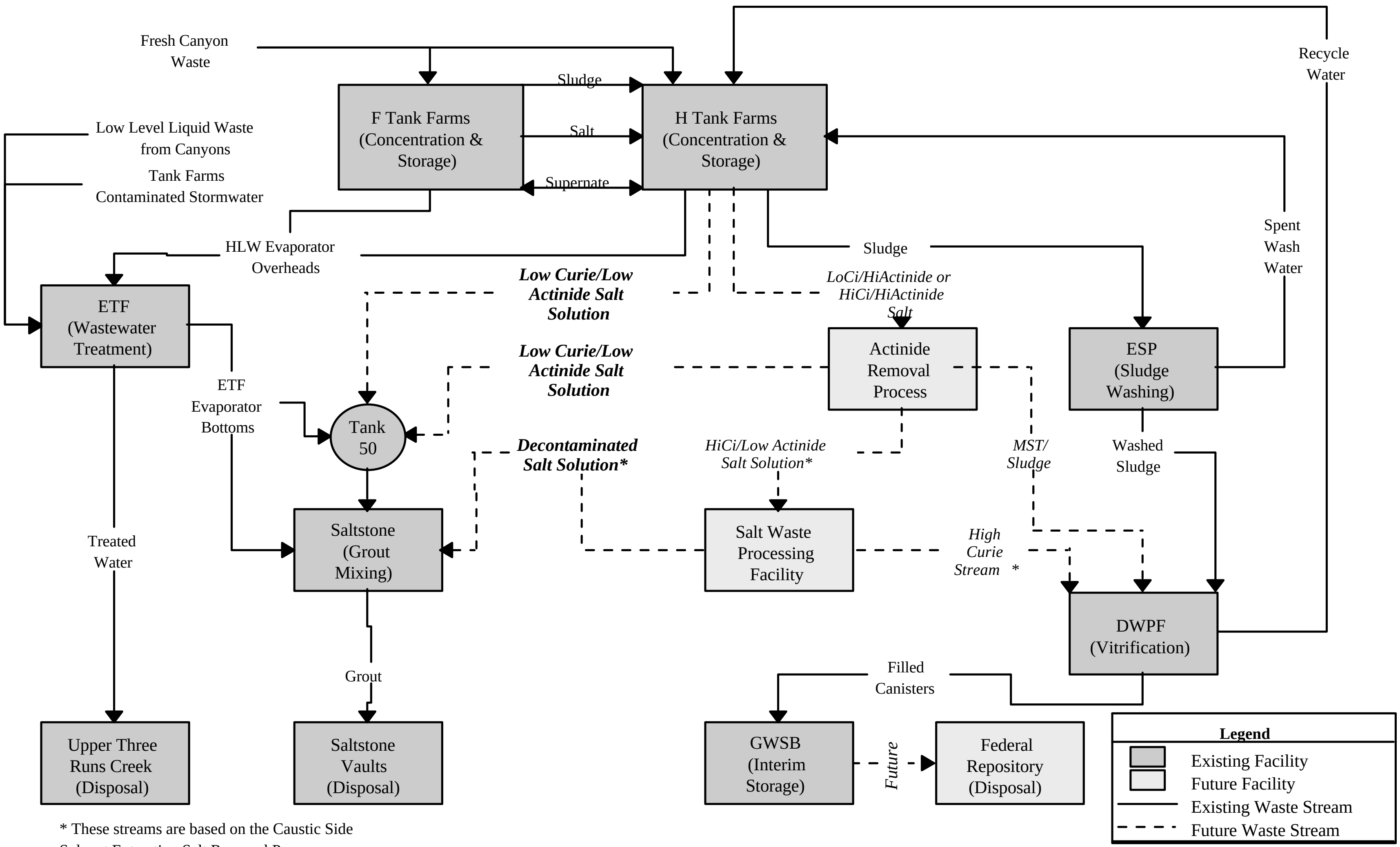
The SRS in South Carolina is a 78000-hectare Department of Energy (DOE) complex that has produced nuclear materials for national defense, research, and medical programs since it began operation in 1951. As a waste by-product of this production, there are approximately 140 million liters of liquid, high-level radioactive waste stored on an interim basis in 49 underground waste storage tanks.

The waste stored in SRS tanks is broadly characterized as either “sludge waste” or “salt waste.” Sludge waste is insoluble and settles to the bottom of a waste tank, typically beneath a layer of liquid supernate. Sludge generally contains the radioactive elements strontium, plutonium, and uranium in the form of metal hydroxides. Sludge is only 8% of the SRS waste volume (~11 million liters) but is 55% of the waste radioactivity (220 million curies).

Salt waste is soluble and is dissolved in the liquid. Salt generally contains the radioactive element cesium and trace amounts of other soluble radioactive elements in the form of dissolved salts. Salt waste is 92% of the SRS waste volume (134 million liters) and 45% of waste radioactivity (180 million curies). Salt waste can be further described as being “supernate” (in normal solution), “concentrated supernate” (after evaporation has removed some of the liquid) or “saltcake” (previously dissolved salts, such as sodium nitrate and sodium nitrite, that have now crystallized out of solution). A single waste tank can contain sludge, supernate, and salt cake; although an effort is made to segregate sludge and salt in different tanks.

Continued, long-term storage of these liquid, high-level wastes in underground tanks poses an environmental risk (twelve of the SRS tanks have a waste leakage history). Therefore, the High Level Waste Division (HLWD) at SRS has, since FY96, been removing waste from tanks; pre-treating it; vitrifying it; and pouring the vitrified waste into canisters for long-term disposal. From 1996 through 2002, 1330 canisters of waste have been vitrified. The canisters vitrified to date have contained sludge waste only.

Figure 1 shows the HLW System flowsheet.



* These streams are based on the Caustic Side Solvent Extraction Salt Removal Process

Figure 1 - Simplified HLW System Flowsheet

Waste made resulting from the production of nuclear materials is neutralized (pH > 12) and transferred from separations facilities ("canyons") into storage tanks. After aging to allow insoluble metal oxides and hydroxides (primarily, of iron, aluminum, sodium, silicon and manganese) to settle, the clear liquor ("supernate") is decanted and evaporated for volume reduction. Evaporator overheads are treated at the Effluent Treatment Facility (ETF) and discharged to the environment.

The settled material forms sludge which is transferred to the Extended Sludge Processing Facility (ESP) to be washed to reduce the level of alkali and to remove anions which interfere with subsequent processing. The wash water is evaporated. The washed sludge is transferred to the Defense Waste Processing Facility (DWPF) where it is treated to adjust the metal oxidation states for glass production and to remove mercury. Glass formers ("frit") are added and the frit-sludge slurry is fed to a large melter. The resulting glass is poured into stainless steel canisters, which are welded close and stored. The operation of DWPF and disposition of sludge have been thoroughly reviewed elsewhere [1] and will not be discussed further.

This paper deals with treating and immobilizing salt waste.

Treatment of Salt Waste

Water-soluble waste is stored as supernatant liquor ("supernate") or as a solid mass produced by evaporation of supernate ("salt" or "salt cake"). SRS now stores 70 million liters supernate and 60 million liters saltcake. Taken together, these materials are "salt waste". When reconstituted as liquid feed for processing at 6.44 M sodium, the volume requiring treatment will be 316 million liters.

The source of most of the activity in this waste is cesium-137 (and its short lived daughter barium-137m). Slightly soluble radionuclides, such as strontium and TRU are also present. In the baseline process, the feed is treated with solid monosodium titanate slurry (MST) to sorb strontium and TRU radionuclides and is filtered, removing trace sludge particles and the MST. Cesium is removed from the clarified feed by solvent extraction. These processes were to be performed in a large facility, built specifically for that mission. Figure 2 shows this baseline concept.

The treated waste is transferred to Z-Area for disposal as “Saltstone” – a solid wasteform of salt solution, slag, flyash and cement. The saltstone is cast into large vaults, each capable of disposing of about 23 million liters each of the wastewater.

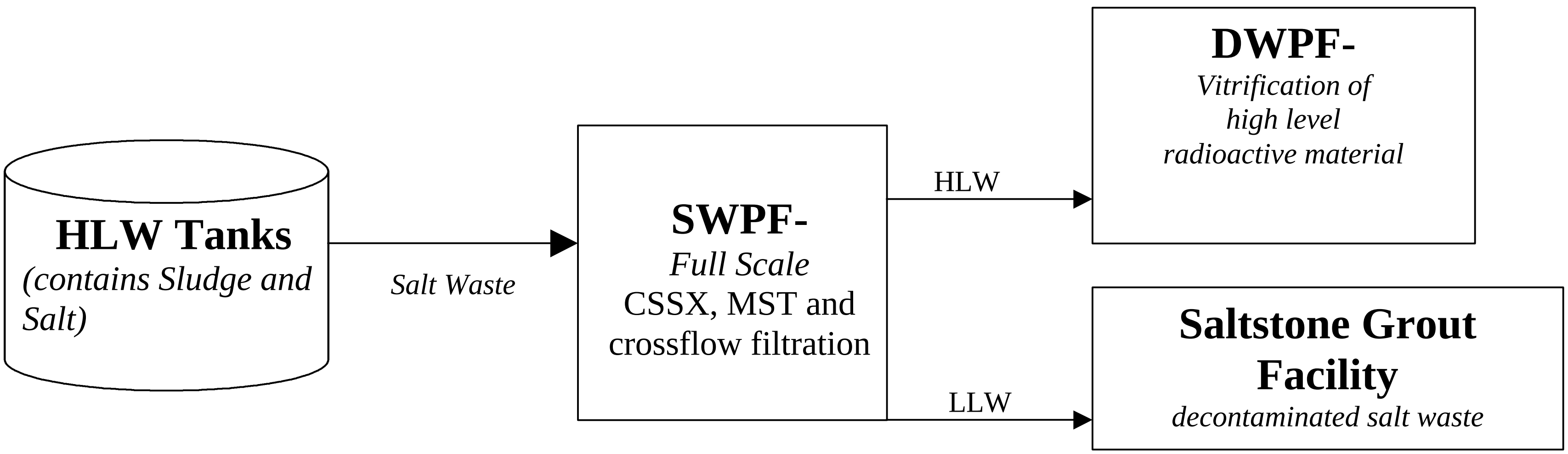


Figure 2 - Baseline Flowsheet

Due to the schedule and cost required to implement the baseline approach, SRS has developed a more cost-effective approach for earlier waste disposition and risk reduction. Using a tailored approach, Salt Waste is treated by three different processes, each appropriate to the type and level of radioactivity in the feeds The Low Curie Salt (LCS) process will be used for Salt Waste low in cesium, strontium and TRU radionuclides. An Actinide Removal Process (ARP) will be used for waste which is low in Cesium but higher in TRU concentrations. The baseline process of actinide removal followed by solvent extraction is used for the concentrated supernates which are high concentrations of both TRU and cesium.

- The LCS waste will be segregated from high curie salt supernate by draining the supernate and interstitial salt solution from the saltcake. Because cesium is soluble, removing 70 to 90 percent of the interstitial salt solution removes an equivalent percent of the cesium and other soluble radionuclides. The remaining saltcake (mostly sodium nitrate and other none radioactive salts) in the tanks will then be dissolved.   Dissolved saltcakes which meet the acceptance criteria of the Saltstone grout facility, will be stabilized and disposed of in the Saltstone Vaults. Dissolved saltcake which do not meet the acceptance criteria will be returned and either treated by the Actinide Removal Process and /or the solvent extraction process.
- Following removal of the interstitial supernate, some dissolved saltcakes may have strontium and actinide levels which are too high for feed to the Saltstone grout facility. Savannah River is modifying the existing building 512-S facilities to perform actinide removal with monosodium titanate and crossflow filtration. This is the Actinide Removal Process.
- The remaining Salt Waste, which contains higher cesium and often higher TRU concentrations than Salt Waste treated as LCS or ARP feed, will be processed in a Salt Waste Processing Facility (SWPF) using sorption with MST to remove strontium and TRU radionuclides, followed by solvent extraction to remove cesium. This facility is planned as a 15% baseline facility.

Figure 3 shows the accelerated, reduced risk concept.

Goal is to reduce the amount of material processed in SWPF and DWPF thus accelerating the schedule and reducing the costs

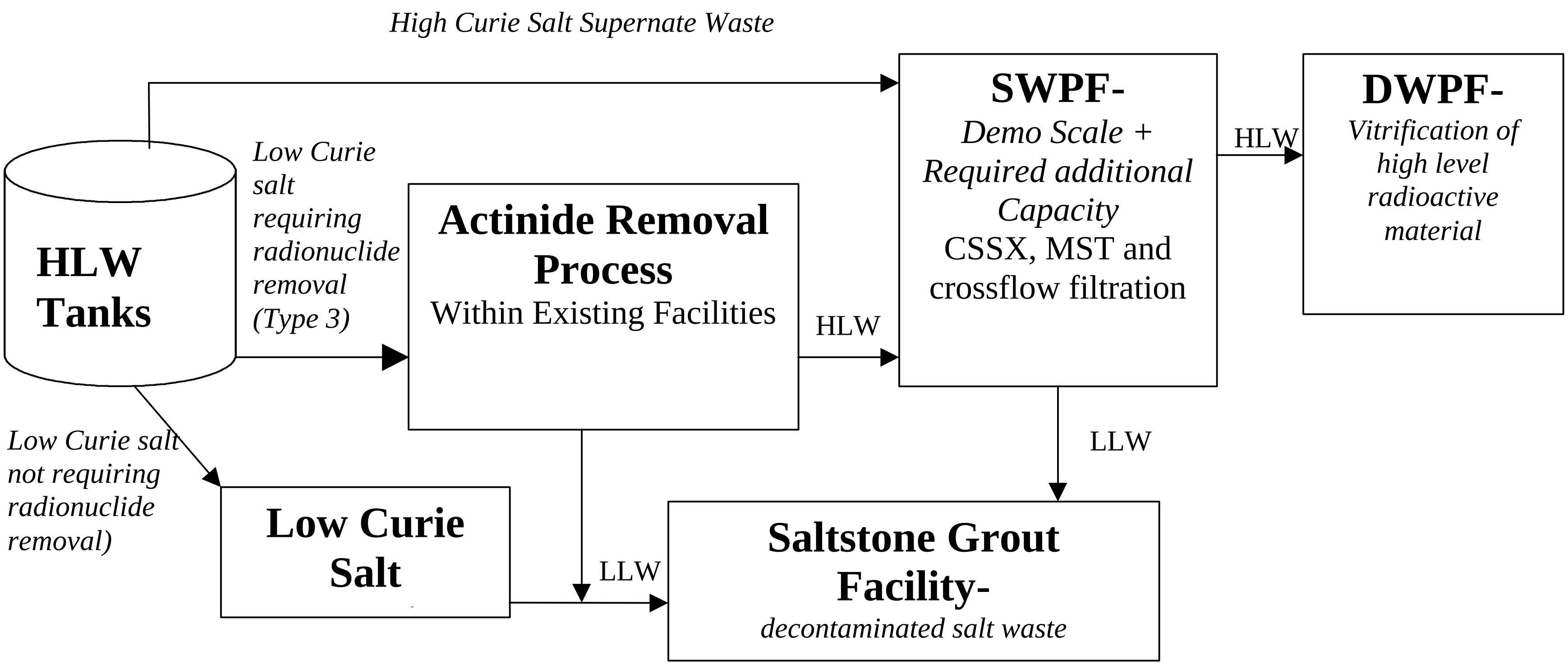


Figure 3 - Accelerated Approach

As shown in figure 3 above the salt waste streams are split into:

- Low curie salt – the low curie path will send the decintaminated salt solution to Saltstone if it meets the waste acceptance criteria (WAC) requirements.
- Low curie salt with higher actinide content – ARP will send a decontaminated salt stream to Saltstone and a monosodium titanate (MST) actinide stream to the Defense Waste Processing Facility (DWPF).
- High curie salt – the high curie salt will be processed in a CSSX Salt Waste Processing Facility (SWPF). The SWPF will send a decontaminated salt stream to Saltstone, an MST actinide stream to DWPF, and an acidified cesium stream to DWPF.

Table I shows the estimated volumes sent through each of these processes.

Table I. Projected Volumes Fed to Salt Processing Alternatives

FY	Low Curie Salt, liters	Actinide Removal Program (ARP), liters	Low Curie Salt+ ARP, liters	SWPF Feed, liters		Total Salt Solution, liters
02	-	-	-	-		-
03	7570000		7570000			7570000
04	15140000		15140000			15140000
05	19310000	1510000	20820000			20820000
06	17040000	3790000	20820000			20820000
07	14010000	7570000	21580000			21580000
08	12490000	8330000	20820000			20820000
09	4160000	8330000	12490000	8330000		20820000
10	3030000	8330000	11360000	10220000		21580000
11	3030000	8330000	11360000	10220000		21580000
12	1890000	9840000	11740000	10220000		21960000
13	1510000	9840000	11360000	10220000		21580000
14		9840000	9840000	10600000		20440000
15		9840000	9840000	10600000		20440000
16		9840000	9840000	10600000		20440000
17		9470000	9470000	10600000		20070000
18				10600000		10600000
19				9470000		9470000
Totals	99190000	104870000	204070000	111690000		315750000

* 100% design flowrate is defined as 22.7 ML/year at 6.44 [Na]

The proposed processing strategy also will reduce the time required to process the salt waste. The end date for salt waste processing is FY2027 for the baseline process. The proposed strategy accelerates the schedule completion to FY2019. Life cycle cost savings could be in excess of $5.4 billion [2].

Regulatory [3]

Waste disposed at Saltstone are to be controlled under DOE Order 435.1, Radioactive Waste Management as Waste Incidental to Reprocessing (WIR). The following WIR requirements must be met:

- Remove key radionuclides to the maximum extent technically and economically possible
- Comply with Performance Objectives comparable to Nuclear Regulatory Commission 10 CFR 61 Subpart C (general requirements for land disposal).

- Produce a solid waste form meeting concentration limits in 10 CFR 61.55 (waste classification for near surface disposal) for NRC Class C Low Level Waste.

As much as 95% of the Cs-137 can be removed from the F- and H- Tank Farm HLW processed using the accelerated waste removal program. The incremental cost to remove the remainder per 1 million gallons of saltcake is $87 million. Reference [3] demonstrated that Processing all the salt waste through a full scale SWPF, versus using the risk based accelerated approach would result in removal of only about 5% of the total activity sent to Saltstone. This is at a cost per gallon six time that of processing through ARP or LCS facilities. Therefore, the key radionuclides will be removed to the extent economically and technically practical.

The saltstone waste form will be assured to meet 10CFR61 and DOE Order 435.1 by Waste Acceptance Criteria developed for transfer of feed to Saltstone. The WAC will be written to establish operational limits within the applicable Performance Objective and NRC Criteria. A Performance Assessment demonstrating that the performance objectives are met for Saltstone was completed in 1992. This has been updated by a Special Analysis [4]. The Special Analysis considers requirements from DOE Order 435.1 and the Performance Objectives of 10 CFR 61, Subpart C. The requirements are met, if the Saltstone vault intrusion barrier is increased from 1.15 m to 1.5 m.

Currently, Z-Area is operating as a wastewater treatment facility followed by an industrial solid waste landfill, both licensed by the South Carolina Department of Health and Environmental Control (SCDHEC). The waste currently meets NRC Class A limits. However, the waste form and its storage vaults are suitable for NRC Class C waste. A Permit modification package is pending with SCDHEC to increase the radionuclide content to Class C. Chemical Components and concentrations remain essentially the same

<u>Program Status</u>

SWPF – DOE Awarded two contracts September 2002 for the preparation of conceptual design, which will be completed in early FY’04. DOE will select one of these contractors to complete the design, procure, build and commission the SWPF. Sorption of TRU and Sr on MST, followed by caustic-side solvent extraction is the flowsheet basis.

LCS – Interstitial liquid is being drained out of the saltcake in Tank 41H. As of January 2003 about half of the liquid has been removed, at a rate somewhat slower than expected. External radiation measurements are decreasing in intensity as the liquid is being removed. These are in qualitative agreement with expected cesium removal. This is the first tank of up to 8 HLW tanks which can be treated using this method.

ARP - The ARP is progressing towards a start target date of January of '04. Existing facilities are being used to the extent possible. The control system and process equipment are being installed. In addition to physical modifications, training, procedures preparation, cold runs, and other activities are scheduled to be completed in 2003. ARP will process waste from up to 12 tanks.

Conclusions

DOE and WSRC are developing an alternative, more cost effective path to dispose of the water soluble HLW (“salt”). Salt Waste is treated by three different processes, each appropriate to the type and level of radioactivity in the feeds. This path is within the existing regulatory framework. Work is underway for all three processes.

Successful implementation would avoid the life cycle cost of building the full scale baseline facility and avoid the HLW operational cost for 8 years ($5.4 billion).

References:

1. S. L Marra , The Savannah River Site Accelerated Clean-Up Mission: Sludge, WSRC-MS-2003-00053, December 2002
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3. J. K. W. Dunaway and J. R. Sessions, Waste Incidental to Reprocessing Evaluation for Disposing Saltcake to Saltstone, HLW-SDT-2001-00281, Rev 1., February 2002
4. E. L. Wilhite *et al*, Special Analysis: Revaluation of the Inadvertent Intruder, Groundwater, Air and Radon Analyses for the Saltstone Disposal Facility, WSRC-TR-2002-00456